

Occupational Exposure Management at Nuclear Power Plants

**Third ISOE European Workshop
Portoroz, Slovenia
17-19 April 2002**

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

© OECD 2003

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14 December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

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, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, 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.

© OECD 2003

Permission to reproduce a portion of this work for non-commercial purposes or classroom use should be obtained through the Centre français d'exploitation du droit de copie (CCF), 20, rue des Grands-Augustins, 75006 Paris, France, Tel. (33-1) 44 07 47 70, Fax (33-1) 46 34 67 19, for every country except the United States. In the United States permission should be obtained through the Copyright Clearance Center, Customer Service, (508)750-8400, 222 Rosewood Drive, Danvers, MA 01923, USA, or CCC Online: <http://www.copyright.com/>. All other applications for permission to reproduce or translate all or part of this book should be made to OECD Publications, 2, rue André-Pascal, 75775 Paris Cedex 16, France.

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.

The workshop objectives are:

- To provide a large forum to exchange information and experience on occupational radiation exposure issues at nuclear power plants.
- To allow vendors to present their recent experiences and current technology in the radiation protection area.

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' underlying 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 Third ISOE European Workshop on Occupational Exposure at Nuclear Power Plants was organised in April 2002, in Portoroz, Slovenia, and attended by 130 participants from 26 countries. Participants represented utilities, regulatory bodies and contractors. The IAEA supported participants from Central and Eastern European countries as well as from Asia. The workshop included 35 oral presentations and eight poster presentations.

The success of this workshop was largely due to the significant organisational support provided by the Krsko nuclear power plant and the Slovenian regulatory body.

TABLE OF CONTENTS

Foreword	3
Introductory Session	9
<i>S. Deme</i> Space Radiation Dosimetry	11
Session 1	21
<i>S. Mundigl and M. Gustafsson</i> Information System on Occupational Exposure – Ten Years of Experience	23
<i>C. Lefaure and P. Crouail</i> Implementation of the Basic Safety Standards in the Regulations of European Countries	35
<i>M. Gustafsson and K. Mrabit</i> Implementation of the International Basic Safety Standards in IAEA Member States: A Unique Experience	47
<i>W. Pfeffer and G. Weimer</i> The New German Radiation Protection Ordinance	55
<i>W. Bines</i> Implementation of the Basic Safety Standards Directive in the UK	75
<i>B. Breznik, M. Križman, D. Škrk and R. Tavzes</i> Law on Protection Against Ionising Radiation and Nuclear Safety in Slovenia	83
<i>D. Viktory</i> Radiation Protection Regulations in Slovakia and Application of BSS and EC Council Directives Provisions	91
<i>G. Klevinskas and A. Mastauskas</i> Transposition of Requirements Set Out in the Basic Safety Standards for Nuclear Facilities in Lithuania	97
<i>J. Lebeau</i> French Government Regulation – New Enactment on Optimised Radiation Protection of Workers Exposed to Ionising Radiation – Implementation at a French Nuclear Power Plant Belonging to the EDF Group	105

Session 2	109
<i>K. Ohr, E. Anderson and D. Barker</i> Quad Cities Unit One Cobalt-60 Experience	111
<i>M. Berger, J.L. Bretelle and A. Rocher</i> Impact of Main Radiological Pollutants on Contamination Risks (ALARA) Optimisation of Physico-chemical Environment and Retention Techniques During Operation and Shutdown	117
<i>E. Fredriksson, R. Ivars, A Rosengren and G. Granath</i> Fuel Decontamination at Ringhals 1 with the New Decontamination Process ICEDEC™	129
<i>I. Terry and B. Breznik</i> Steam Generator Replacement from ALARA Aspects.....	137
<i>G. Volent and I. Metzger</i> Applying ALARA for Replacements of Steam Generator Feedwater Distribution Pipes at Paks NPP	145
<i>L. Vermeulen</i> Replacement of Neutron Absorbers in the Spent Fuel Pool Storage Facility of the Tihange 3: An ALARA Approach.....	149
<i>J.P. Degrange, C. Benhamou, L. D'Ascenzo, A. Leroy and E. Vrignaud</i> Analysis of the Doses Associated with the Spent Fuel Shipments from the French NPPs: Are they ALARA?	159
<i>V. Atoyán and A. Muradyan</i> Doses of the Staff During the Spent Fuel Assemblies Transportation and Storage in NUHMOS 56V Concrete System	173
Session 3	187
<i>V. Pletniov and G. Klevinskas</i> Enhancement of Radiological Safety Culture Within Ignalina NPP and Contractors	189
<i>M. Gonin, C. Bailloeuil, A. Pétrequin and M. Teissier</i> About the Interest of an Agreement for a European Passport for Workers in European Nuclear Fuel Cycle Installations.....	197
<i>M. Andersson, A. Holmqvist and J. Möller</i> A Swedish Dose Passport – Contractors Point of View.....	201
<i>A. Weeks, M. Pottinger and P. Clarke</i> The BNFL Legal Electronic Dosimetry Service.....	207
<i>S. Morris</i> CDRL – Company Dose Restriction Level.....	215

<i>W. Wood and D. Miller</i> Commonality Initiatives in US Nuclear Power Plants to Improve Radiation Protection Culture and Workers Efficiency.....	223
<i>J-L. Fenolland and E. Laporte</i> Industrial Accidents in Radiological Controlled Areas: The Importance of Radiation Protection in the Organisation of the Emergency Aid.....	235
<i>R.S. Schofield</i> Health Physics Self-assessment and the Nuclear Regulatory Oversight Process at a Nuclear Power Plant.....	239
Session 4.....	253
<i>R.L. Doty and R.C. Kessler</i> Discrete Radioactive Particle Issues Associated with a Fuel Pool Cleanout	255
<i>A. Roupioz and J.F. Labouglie</i> Controlling the Alpha Risk in the EDF Facilities	261
<i>G. Valtchev, L. Dimitrov and M. Neshkova</i> Dose Assessments from Whole-body Measurements – The Approach at NPP Kozloduy.....	265
<i>S. Jaroslav</i> Contamination Control at the Exit from Radiation Controlled Area and NPP Site at Bohunice NPP.....	271
<i>F. Schartmann and S. Thierfeldt</i> Radiological Assessment of Radioactive Contamination on Private Clothing	279
<i>M. Meyer</i> Radioactive Contamination on Private Clothing.....	289
<i>M.P. Lunn and G.A. Renn</i> Practical Limitations of Monitoring Personal Contamination at a PWR	295
Posters.....	303
<i>M. Černilogar Radež, H. Janžekovič and M. Križman</i> Dose Trends Analysis of the Krško NPP	305
<i>O. Guzmán, T. Labarta, J.J. Ontesinos, M.L. Rosales, M.J. Muñoz and I. Amor</i> Practical Implementation of the 96/29 Euratom Directive to the Radiation Protection Programme of Spanish Nuclear Power Plants	311
<i>S. Hübner and W. Wahl</i> Present and Future Aspects of Harmonisation in Official and Operational Personal Dosimetry	315

<i>J. Koc and K. Petrová</i>	
The Exercise of the ALARA Principle at the Installation of the Primary Neutron Sources into the Fuel Assemblies at the Temelín NPP	321
<i>T. Lisova and V. Novikov</i>	
The Implementation of the Radiological Safety Standards in Ukraine.....	327
<i>I. Villanueva, A. Hernandez, M.J. Muñoz and I. Amor</i>	
Radiation Passbook and Training Model for Outside Workers in Spain	333
List of Participants	339

INTRODUCTORY SESSION

SPACE RADIATION DOSIMETRY

S. Deme

KFKI Atomic Energy Research Institute, Budapest, Hungary

Abstract

The radiation environment in low earth orbital flights is complex. The paper summarises the component of the space radiation, describes the dosimetric methods usually applied for space radiation dosimetry, and gives the typical figures of the measured results. The last part deals with dose limitation system of astronauts.

Introduction

Although partly protected from galactic and solar cosmic radiation by the Earth's magnetosphere in **Low Earth Orbit** (LEO) astronauts exposure levels during long-term missions (90 days to 180 days) by far exceed with exposures of up to more than 100 mSv the annual exposure limits set for workers in the nuclear industry, but are still below the yearly exposure limits of 500 mSv for NASA astronauts. During solar particle events the short term limits (300 mSv) may be approached or even exceeded, which certainly would not be life threatening, but raise the probability for developing cancer later in life considerably.

In the **interplanetary space**, outside the Earth's magnetic field even relatively benign Solar Particle Events (SPEs) can produce 1 Sv skin-absorbed doses while the huge SPE observed in August 1972 for example would have resulted in a skin-absorbed dose of 26 Sv behind a shielding of 0.5 g/cm^2 and would have proved lethal had there been any Extravehicular Activity (EVA) during this event. Although new rocket technologies could reduce astronauts' total exposure to space radiation during a human Mars mission, the time required for the mission which is now in the order of years remains still about half of this time. Therefore mission planners will need to consider a variety of countermeasures for the crew members including physical protection (e.g. shelters), active protection (e.g. magnetic protection), pharmacological protection, local protection (extra protection for critical areas of the body) etc. With full knowledge of these facts, accurate personal dose measurement will become increasingly important during human missions to Mars.

Space Radiation

Charged particles

Usual grouping of the charged particle components of the space radiation (Reitz *et al.*, 1989):

- geomagnetically trapped radiation;
- solar-particle radiation;
- galactic cosmic rays.

Trapped radiation

As a result of the interaction of particles coming from outer space or the Sun with the geomagnetic field, there are two belts of trapped radiation, the 'Van Allen belts', surrounding the Earth as rings in the plane of the geomagnetic equator. Mostly electrons and protons are present in both belts. These particles gyrate and bounce along magnetic-field lines and are reflected back and forth between the two poles, acting as mirror. At the same time, because of their charge, electrons drift eastward, while protons and heavy ions drift westward. Electrons reach energies of up to 7 MeV and protons up to 600 MeV.

For the majority of space missions in lower Earth orbits, protons make the dominant contribution to the radiation burden inside space vehicles. At lower shielding (in the case of EVA – extravehicular activity) the total absorbed dose will be dominated by the electron contribution. Of special importance for low Earth orbits is the so-called 'South Atlantic Anomaly' (SAA), where the radiation belt reaches down to altitudes of 200 km. This behaviour reflects the displacement of the axis of the geomagnetic (dipole) field with respect to the axis of the geoid, with a corresponding distortion of the magnetic field. This region accounts for the dominant fraction of total exposure in ISS, although traversing the anomaly takes less than about 15 minutes and occupies less than 10% of the total time in orbit.

Solar-particle radiation

High-energy solar protons and heavy ions, which are among the most severe hazards for manned space flights, are emitted sporadically during solar-particle events (flares). Flares are observed mainly during the solar-maximum phase. The spectra of solar-flare protons have a large variability in absolute intensity as well as in shape. Doses up to 10 Gy can be reached. On mission outside the magnetosphere, radiation shelters of sufficient thickness must be provided.

Galactic cosmic rays

Galactic cosmic rays (GCR) are charged particles that originate from sources beyond solar system. The GCR spectrum consists of 98% and heavier ions and 2% of electrons and positrons. The ion component is composed of 87% protons, 12% of alpha particles and the remaining 1% heavy ions.

The flux of GCR is affected by the sun's eleven-year cycle. During that period of the solar cycle called Solar Maximum, when solar activity is most intense, the solar wind attenuates a greater flux of the inbound GCR than during Solar Minimum, when solar activity is least intense.

GCR, being composed of charged particles, is also affected by the Earth's magnetic field. Since the geomagnetic field lines are parallel to the Earth's surface around the equator, all but the most energetic particles are deflected away. The geomagnetic field over the North and South Poles points towards the Earth's surface and GCR particles of all energies are funnelled toward the poles at high latitudes. The 51.56° orbit of the ISS is sufficiently highly inclined to receive a substantial exposure from less energetic GCR (Benton and Benton 2001).

Neutrons

High-energy secondary neutrons produced by interactions of high-energy charged particles (from the trapped belts and cosmic rays) contribute a significant fraction of the total dose equivalent in large human spacecraft as the International Space Station (ISS). The two basic components of the neutron radiation are the albedo neutrons emanating from the Earth's atmosphere and the secondary neutrons from the interaction of high-energy space radiation with spacecraft materials. The neutron energy range of interest for radiation risk assessment is 0.1 to at least 200 MeV. Based on both the modelling results and a few measurements covering a portion of the energy range of interest, it was found (USRA 1998) that secondary neutrons contribute a minimum of additional 30% and up to 60% of the dose equivalent rates of charged particles.

Methods to determine dose

Measurements of dose and dose equivalent of charged particles

Because the radiation field in space is a mixture of different particles, which differ also in energy, and varies with time (all solar cycles are different), it is difficult or all but impossible to calculate doses from earlier measurements.

There are three main methods for the determination of the astronaut's radiation exposure:

1. calculation approach;
2. on-line measurements with active devices; and
3. measurements with integrating, passive dosimeters.

Calculation approach

Accumulated dose and dose-rate can be calculated from the information delivered on-line by active devices (tissue equivalent proportional counters, silicon detector systems) used as area dosimeters.

Tissue equivalent proportional counter (TEPC) is simply a spherical or cylindrical detector constructed of tissue-like plastic material and filled with tissue equivalent gas. If operated at low pressure (few percent of atmospheric pressure) this type of instrument allows the simultaneous

determination of the absorbed dose to tissue. The pulse height spectra are usually calibrated in terms of the microdosimetric quantity linear energy (which is related to LET) and can be used to assess the effective quality factor (Q) in complex radiation fields including neutrons, photons and diversity of charged particles. The measured absorbed dose and Q are used to evaluate dose equivalent.

Silicon detectors and detector system can be used as solid state ionisation chamber (case of a single detector) or as a LET spectrometer (case of telescopes). One type of such LET spectrometers is the DOSTEL. The DOSimetry TELescope DOSTEL is based on two identical passivated implanted planar silicon (PIPS) detectors and designed to measure the energy deposit of charged particles (Beaujean 1999). Both detectors have the same thickness (~0.3 mm) and sensitive area (~7 cm²). The distance of 15 mm between the two detectors yields a geometric factor of 1.2 sr for particles arriving from the front when a coincidence in both detectors is required.

On-line measurements with active devices

Active personal dosimeters such as small silicon detectors or small ionisation chambers may be used. Such devices need power and are difficult to design small enough. They are well known due to their application in NPPs.

Measurements with integrating, passive dosimeters

Passive integrating detector systems such as thermoluminescent detectors (TLDs) are commonly used for environmental monitoring and for personal dosimetry. Such TLD measurements need to be supported by spectroscopic information about the high LET part of the radiation field from other instrumentation.

The most known advantages of passive detector systems are their independence of the power supply, small dimension, sensitivity, good stability, wide measuring range, resistance to environmental changes and relatively low cost. Therefore, they are commonly used for long term measurements from several hours up to months and years.

TLDs are perfect for recording absorbed doses from radiation up to a LET of 20 keV/μm. Above this value the efficiency decreases rapidly with increasing LET. The response function of different TL materials as a function of LET has already been determined through a series of calibration. But this knowledge is not sufficient to allow the determination of the absorbed dose and the dose equivalent for the complete radiation field. The dose equivalent is the product of a quality factor (defined as function of LET) and the absorbed dose and is a measure of the radiation exposure of the astronaut. The TLD response and the quality factor can be calculated if TL detectors are supplemented by passive plastic nuclear track detectors (PNTDs) for measurement of LET spectra ≥5 keV/μm in water. TLDs and PNTDs are exposed together and the LET spectrum measurements from PNTDs are used to correct the dose measured in TLDs and, using the corrected dose, to determine dose equivalent.

TLDs are regularly used on board spacecraft but because of the large dimension and big mass of the readers they are typically evaluated only after their return to the ground, in special laboratories. On-ground evaluation has the disadvantage that it results in the dose accumulated since the last read-out i.e. the total dose of the whole flight. Long duration space flights (e.g. on board space stations or at future interplanetary missions) requires time resolved measurements, since this information is needed for radiation risk estimates. A small, portable and space qualified TLD reader

suitable for reading out the TL dosimeters on board provides the possibility to overcome the above-mentioned disadvantage.

Since the end of the seventies KFKI AEKI has developed and manufactured a series of TLD systems named 'Pille' (Butterfly in English) for spacecraft. The system consists of a set of TL dosimeters and a small, compact TLD reader suitable for on-board evaluation of the dosimeters. By means of such a system highly accurate measurements were and are carried out on board the Salyut-6 (Fehér *et al.*, 1981), Salyut-7 (Akotov *et al.*, 1984) and MIR (Deme *et al.*, 1999a, b) Space Stations as well as the Space Shuttle. A new implementation of the system is and will be placed on several segments of the International Space Station (ISS) (Apathy *et al.*, 1999) as the contribution of Hungary to the great international enterprise.

Extended missions call for a measurement of the time profile of the radiation exposure, which cannot be received from PNTD measurements. Their use is also hampered by the fact that PNTDs cannot be read out on-orbit; they need to be returned to the ground for chemical processing and analysis. The solution is to combine the TLD reader with an active LET spectrometer like DOSTEL (Reitz 1998).

Neutron dosimetry

The neutron dosimetry on the ISS requests application of a set of different dosimeters. Concerning this question the Predictions and Measurements of Secondary Neutrons in Space Workshop (USRA 1998) gave the following recommendations.

1. Provide crew personal dosimeters that are sensitive to secondary neutrons in the energy range of interest (0.1 to 200 MeV). At present, the best candidate appears to be CR-39 plastic track detectors. This is a minimum requirement in order to document crew exposure adequately.
2. Develop a proportional counter that would be sensitive only to charged particles and fly it with an existing tissue equivalent proportional counter that is sensitive to both charged particles and neutrons (but cannot distinguish between them) to obtain a measurement of the neutron contribution to the total LET spectrum.
3. Develop and fly Bonner spheres with lead (Pb) and iron (Fe) shields to obtain measurements of the high-energy neutron component.
4. Provide direct-reading active and/or passive dosimeters that are sensitive to neutrons in the energy range of interest. Such a device is needed to allow crewmembers to manage their exposures by the principle of *ALARA* during the flight. The minimum sensitivity for such an instrument should be 100 μ Sv.

Bubble detectors

One of the suitable solutions of the crew personal dosimeter is application of bubble detectors (Ing 1998). The bubble detectors made use of the stored mechanical energy in superheated liquid to amplify the effect of the neutron interactions. It consists of microscopic droplets of superheated liquid – that is, liquid that ought to be in the vapour phase, but is maintained in a liquid phase – dispersed throughout an elastic polymer. When radiation strikes these droplets, the energy from the charged particles triggers the droplets to explode. The resulting gas bubbles, which are visible, are trapped in the elastic medium at the positions of formation to provide a record of

interactions. The number of bubbles is a measure of the neutron dose. The usual range of bubble detectors is 20-200 μSv , the energy range covers 0.4-15 MeV. In general, the tests done by user group show that the bubble detector (model BD-PND: bubble detector – personal neutron dosimeter) has no difficulty meeting the requirements of regulatory agencies for personal neutron dosimetry.

Results of radiation measurement

A review of radiation measurements on both **U.S. Space Shuttle and Mir orbital station** has been made (Badhwar 2000). It shows that

1. The cosmonaut dose varied from a low of 24.3 mGy to a high of 81.8 mGy.
2. The average cosmonaut dose rates, uncorrected for TLD inefficiency and neutron component, varied from 182 $\mu\text{Gy d}^{-1}$ to 397 $\mu\text{Gy d}^{-1}$.
3. During the solar minimum, the quality for GCR varied from 3 to 3.6, trapped from 1.66 to 1.88, and the combined quality factor from 2.14 to 2.51, depending upon the Mir module.
4. Using the quality factor of 2.5 measured in the Core module, where the cosmonauts spent most of their time and assuming the factor applies during solar maximum also, the dose equivalent rates would range from 457 $\mu\text{Sv d}^{-1}$ to 996 $\mu\text{Sv d}^{-1}$.
5. If one corrects the TLD dose rate for their inefficiency at high LET and includes the dose from high energy neutrons, these values could be roughly 25% higher; then the skin dose equivalent rates would range from 571 457 $\mu\text{Sv d}^{-1}$ to 1,246 $\mu\text{Sv d}^{-1}$.
6. The average astronaut dose rate in the Space Shuttle varied from 0.2 mGy to 32.1 mGy, with the highest dose rate of 3,211 $\mu\text{Gy d}^{-1}$ or nearly six times the highest cosmonaut dose rate (468 $\mu\text{Gy d}^{-1}$). This is of course, due to the higher Shuttle flight altitude.
7. Neutron contribute between 15 and 25% of the charged particle dose equivalent but have never been included in astronaut exposures.
8. The east-west asymmetry is a very significant factor in flight with fixed altitude. As such, it would be important for the ISS.

International Space Station

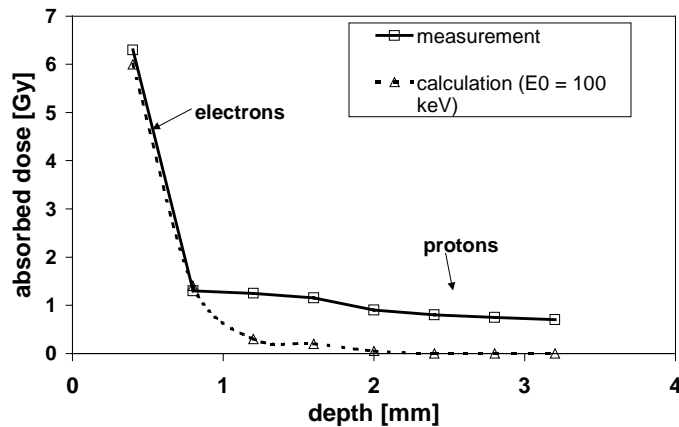
The calculated by Badhwar (Badhwar 2000) dose rate and dose rate equivalent for 400 km of altitude are equal 0.35 mGy d^{-1} and 0.9 mSv d^{-1} .

Gradient of the surface dose

Experiments at the surface of the Mir orbital station were carried out (Schöner *et al.*, 1999). Four stacks of TL-dosimeters of different types were exposed in free space. The stacks contained dosimeters of different thicknesses and were covered with a foil of only 1.26 mg cm^{-2} . Within these stacks the gradient of dose and LET was determined in depths from about 1 mg cm^{-2} to 3 g cm^{-2} . The measured surface dose was in the range of 50 Gy followed by a steep gradient.

Assuming a Maxwellian energy distribution for the cosmic electrons, the initial electron energy can be calculated by Monte Carlo methods to approximately 100 keV. The steep gradient is caused by the electrons absorbed in the dosimeter stack, whereas the cosmic protons penetrate through the material almost unattenuated (Fig. 1).

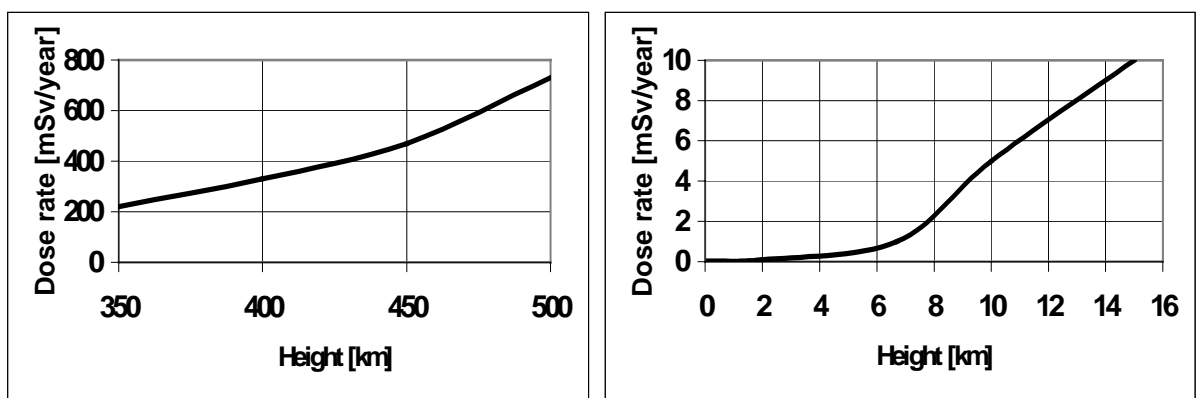
Figure 1. Gradient of dose measured at the surface of space station MIR (Schöner *et al.*, 1999)



The expected average annual dose as a function of flight altitude

For the same circumstances (inclination, wall thickness, orientation, sun activity etc.) the basic component of the dose is the flight altitude. This dependence of the annual dose is given in Figure 2a. For comparison the height dependence of the aeroplane dose is given in Figure 2b.

Figure 2. The expected average annual dose as a function of flight altitude for case of orbital stations (a-left) and aeroplanes (b-right)



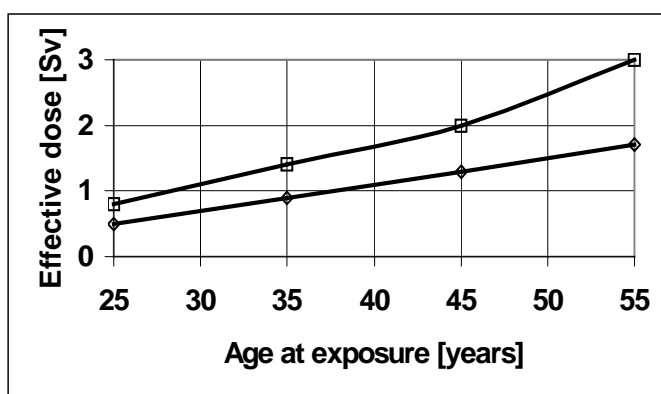
Dose limitation

From the outset of manned exploration of space, it was recognised that exposures of individuals could easily exceed those on Earth unless limiting procedures were adopted. In the low-earth orbit both the inner radiation belt of trapped protons and GCR constitute radiation fields much more intense than on the Earth. Consequently, only limitations on time of exposure could keep exposures to human within reasonable bounds. In addition, solar particle events (SPE) unpredictable in occurrence and intensity could be responsible for large episodic exposures to protons requiring further limitation of exposure conditions.

The new dose limits (Sinclair 2000) for radiation workers correspond to excess lifetime risk of 3% (NCRP) and 4% (ICRP). While astronauts accept the whole variety of flight risks they are taking in mission, there is concern about risks that may occur later in life. A risk no greater than the risk of radiation workers would be acceptable.

The actual recommended values are for 3% excess lifetime (or career) risk are shown in Figure 3.

Figure 3. Career limit (Sv) vs. age at exposure (10 y duration)



References

1. Akatov, Yu. A. *et al.*, Thermoluminescent dose measurements on board Salyut type orbital stations. *Adv. Space Research*, **4**, 77-81, 1984.
2. Apáthy, I., Bodnár, L., Csőke, A., Deme, S. and Héjja, I. An on-board TLD system for dose monitoring on the International Space Station (ISS), *Radiation Protection Dosimetry*, **84**, 321-323, 1999.
3. Beaujean, R., Kopp J. and Reitz, G. Active Dosimetry on Recent Space Flights, *Radiation Protection Dosimetry*, **85**, 223-226, 1999.
4. Badhwar, G. D. Radiation Measurements in Low Earth Orbit: U.S. and Russian Results, *Health Physics*, **79**, 507-514, 2000.
5. Benton, E. R. and Benton, E. V., Space radiation in low-Earth orbit and beyond. *Nucl. Instr. & Meth. B*. 184 pp 255-294; 2001.

6. Deme, S., Reitz, G., Apáthy, I., Héjja, I., Láng, E. and Fehér I, Doses due to the South Atlantic Anomaly during the EUROMIR'95 mission, measured by an on-board TLD system, *Radiation Protection Dosimetry*, **85**, 301-304, 1999a.
7. Deme, S., Apáthy, I., Héjja I., Láng, E. and Fehér, I. Extra dose due to EVA during the NASA4 mission, measured by an on-board TLD system, *Radiation Protection Dosimetry*, **85**, 121-124, 1999b.
8. Fehér, I., Deme S., Szabó, B., Vágvölgyi, J., Szabó, P.P., *et al.*, A new Thermoluminescent Dosimeter System for Space Research, *Adv. Space Res.*, **1**, 61-66, 1981.
9. Ing, H, Neutron Measurements Using Bubble Detectors–Terrestrial and Space in *USRA* 1998.
10. Reitz, G., Facius, R. and Bücker, H. Radiation Biology in *Life Sciences Research in Space*, ESA SP-1105, 1989.
11. Reitz, G., Beaujean, R., Heilmann, C., Kopp, J., Leicher, M., and Strauch, K., Results of dosimetric measurements in space missions, *Adv. Space Res.*, **22**, 495-500, 1998.
12. Schöner, W., Hajek, M., Noll, M., Ebner, R., Vana, N., Fugger, M., Akatov, Y., Shurshakov, V. Measurement of the depth dose– and LET distribution at the surface and inside of space station MIR, in *Proceedings of IRPA Reginal Congress on Radiation Protection in Central Europe*, Budapest, ISBN 96-380-5188-4, 570-575, 1999.
13. Sinclair, W. K., Dose limits for astronauts, *Health Physics*, **79**, 585-590, 2000.
14. Universities Space Research Association. (USRA) *Proceedings of Workshop Predictions and Measurements of Secondary Neutrons in Space*, Houston, 1998.

SESSION 1

INFORMATION SYSTEM ON OCCUPATIONAL EXPOSURE – TEN YEARS OF EXPERIENCE

S. Mundigl*, **M. Gustafsson****

*NEA ISOE Joint Secretary, Issy-les Moulineaux, France

**IAEA ISOE Joint Secretary, IAEA, Vienna, Austria

1. Introduction

During the 1980s, radiation protection experts in the nuclear industry, at operating organizations and at regulatory authorities were faced with new challenges in the management of worker protection at nuclear power plants. The main issue was the growing pressure to put into practice the conceptual approach of optimization of protection, which at that time was becoming one of the cornerstones of international radiation protection standards. This almost naturally generated a feeling that worldwide progress in applying the optimization principle to the control and reduction of worker exposures could be achieved if the variety of managerial and operational approaches adopted in different nuclear power plants and different countries were pooled, exchanged and compared in an organized way.

But this would require a mechanism to exchange and review experience between health physicists. The idea was raised to create an international database and a network of contacts and assistance, with the aim of establishing a bridge between regulators and operators in areas of common interest by involving regulatory authorities in discussions on the implementation of the “as low as reasonably achievable” (ALARA) principle based on operational information. This idea proved to be successful, as is demonstrated by today’s participation in ISOE (the Information System on Occupational Exposure) of regulatory authorities from 25 countries.

ISOE was created in 1992 to provide a forum for radiation protection experts from both operating organizations 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. The ISOE programme is managed by a Steering Group, whose chairman is selected among representatives from the participating utilities.

The ISOE programme offers a variety of products in the occupational radiation exposure arena, such as:

- The world’s largest database on occupational radiation exposure from nuclear power plants. As of December 2001, the ISOE database includes information on occupational radiation exposure levels and trends at 460 reactor units (406 in operation and 54 in various phases of decommissioning), operated by 73 utilities in 29 countries. This

database thus covers some 93% of the total number (438) of power reactors in commercial operation throughout the world.

- A yearly analysis of dose trends and an overview of current developments, through ISOE Annual Reports. The Annual Reports [1] summarise recent information on levels and trends of average annual collective dose at the reactors covered by the database, provide special data analyses and dose studies, outage experience reports, summaries of ISOE workshops and symposia, as well as information on principal events in ISOE participating countries.
- Detailed studies and analyses, as well as information on current issues in operational radiation protection, through ISOE Information Sheets. Dosimetric and other data from nuclear power plants provide an ideal basis for studies on dose related to certain jobs and tasks, such as refuelling, steam generator replacement, insulation work, etc. These studies are published as ISOE Information Sheets and distributed to ISOE participants.
- A system for rapid communication of radiation protection information, such as effective dose reduction approaches, effective decontamination procedures and implementation of work management principles. Anytime an operating organization wishes to share experience on good practices, radiological problems or other technical issues, the ISOE network may be used to request or send information through the Email system. This allows rapid responses and interaction between interested participants.
- A forum for discussing occupational radiation exposure management issues through ISOE workshops and symposia. Each year, an international workshop or symposium on occupational radiation exposure management at nuclear power plants is organised, in turn, in Europe and North America. The objective of these workshops and symposia is to provide a forum for radiation protection professionals from the nuclear industry, operating organizations and regulatory authorities to exchange information on practical experience on occupational radiation exposure issues in nuclear power plants.

A part of the above-mentioned components of the Programme are reserved for sole use by the participating operating organizations, which can thus dispose of a closed network for information exchange on particular operational experiences.

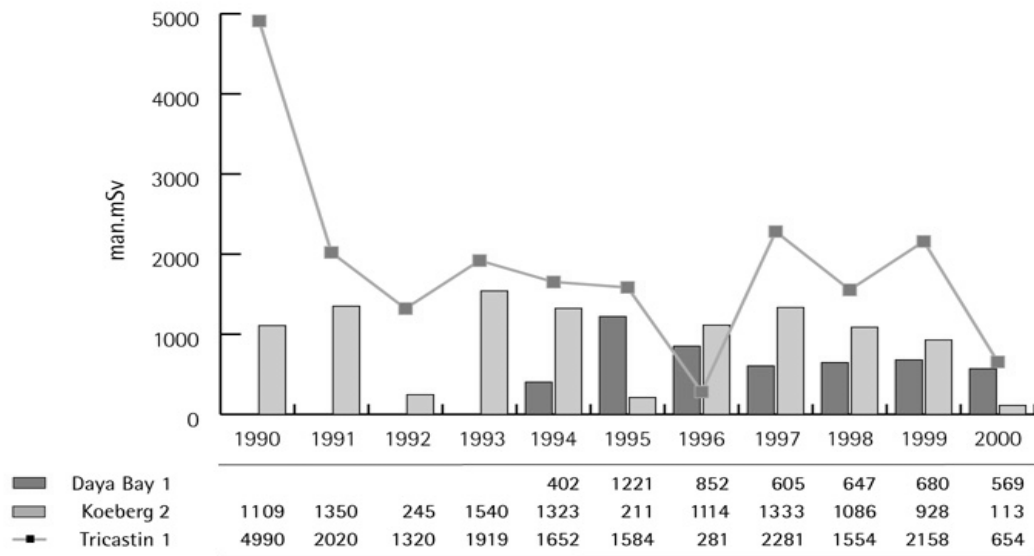
This paper is based on a report [2], published in 2002, summarizing the experience gained from ten years of developing ISOE. The full report contains, in addition to further details, comments from participants on their experience with the system.

2. Radiation protection professionals benefit from ISOE

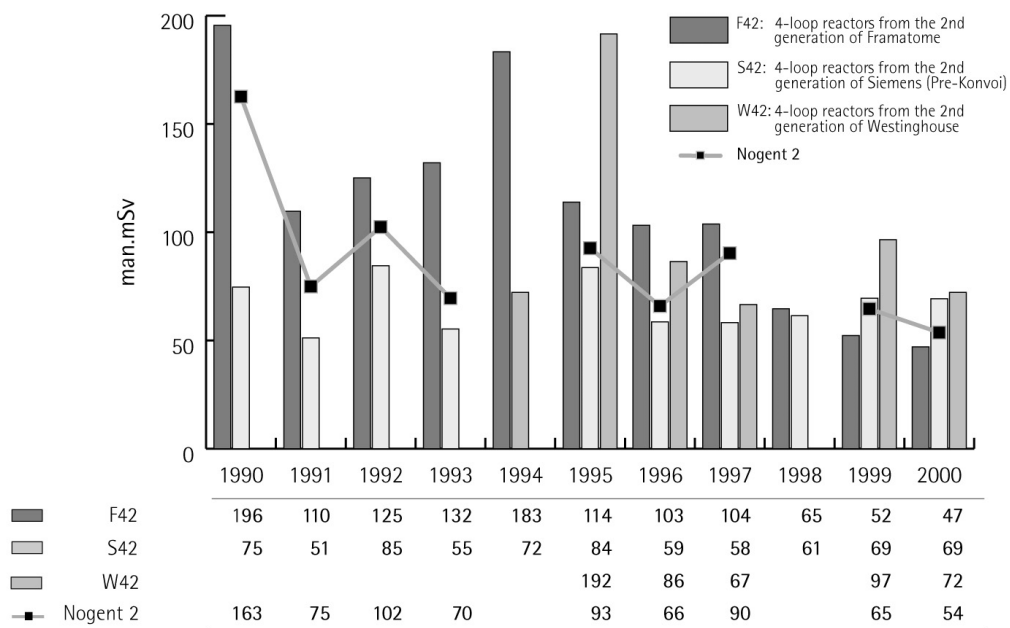
2.1 *Benchmarking analysis*

The ISOE database forms an excellent basis for studies and comparisons of occupational radiation exposure data between nuclear power plants in various countries or even within the same country. To improve the significance and usefulness of these studies, comparative analyses of data from reactors having similar characteristics can be made. For this purpose, “sister unit groups” have been defined within the ISOE database, each containing reactor units of comparable type and design. Except for gas cooled reactors, each reactor included in the ISOE database has been assigned to a sister unit group.

Annual dose benchmarking for Tricastin 1 between 1990 and 2000



Annual dose benchmarking on the job "refuelling" for Nogent 2 between 1990 and 2000



Using the ISOE software,¹ participants are able to generate pre-defined benchmarking tables and graphs. They can create their own comparisons with other units, within the relevant sister unit group and/or in other sister unit groups. The benchmarking analysis is available at various levels, such as annual collective dose and dose per job (e.g. refuelling, steam generator primary side, etc.). Examples are given in Figures 1 and 2.

For a more detailed understanding of the results, participants can directly contact the responsible counterparts in other nuclear power plants by using the contact information available within the ISOE database.

2.2 *Experience exchange*

The communication network available to participants, using modern technology for real-time information exchange, is one of the most useful features of ISOE. ISOE participants can use their respective Technical Centre to obtain information and advice on specific radiological problems, radiation protection techniques, procedures of work, and more. Each ISOE Technical Centre investigates questions raised by a participant, by contacting other ISOE participants directly or through the other ISOE Technical Centres. The resulting information is passed on to the questioner. In cases of general interest, a summary is published as an ISOE Information Sheet.

For the above purpose, an E-mail system has been installed at the NEA Secretariat. This system also allows ISOE participants to exchange reports, questions and other information electronically with all other ISOE participants (utilities or authorities only, or both).

ISOE expert groups can be established to conduct specific studies based on the needs of the participants. For example, an expert group was created to quantify the impact of work management on occupational radiation exposure. The report [3] generated by this group was widely distributed and translated into several languages.

As already noted, several types of documents are made available to ISOE participants. These include the following:

- ISOE Annual Reports [1] presenting the evolution of occupational radiation exposure in nuclear power plants, as well as information on principal relevant events in the ISOE participating countries;
- ISOE Information Sheets (with “general distribution” to all participants or “limited circulation” to utilities only);
- reports issued by expert groups.

Additional exchanges of experience take place during the annual Steering Group meetings. The ISOE Steering Group consists of representatives from operating organizations and regulatory bodies who, besides deliberating on ISOE management issues, review current developments and national trends in the operation and regulations of the nuclear industry, from a radiation protection expert’s perspective.

1. ISOE provides participants with software packages, including the ISOE database and the input module (ISOEDAT) and the interface programme containing pre-defined analyses (MADRAS).

2.3 *Symposia and workshops*

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.

The objectives of these meetings include the following:

- To provide a large forum to exchange information and experience on occupational radiation exposure issues at nuclear power plants.
- To allow vendors to present their recent experiences and current technology in the radiation protection area.

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.

Further information exchange is accomplished by having the three best papers from each workshop presented at an alternate workshop. These papers and additional information are available on the European Technical Centre website (<http://isoe.cepn.asso.fr/>) and the North American Technical Centre website (<http://hps.ne.uiuc.edu>). Non-participating individuals and institutions have access to these websites.

2.4 *Expert group on work management*

The ISOE Steering Group published an expert group report on Work Management in the Nuclear Power Industry in 1997. This was one of the first ISOE products that documented good radiological work management practices aimed at reducing occupational doses.

The preparation of the report [3] started in 1995 with the creation of an ISOE expert group of radiation protection managers from eight countries, including Canada, Finland, France, Germany, Sweden, Switzerland, the United Kingdom and the United States. The expert group was chaired by the United States.

The contents of the report cover work planning, including scheduling and training, implementation and feedback.

Feedback from ISOE participating utilities on the report has been exceptionally positive. For example, reported applications of this document by US participating utilities include:

- use of the report's outline and text as an ALARA assessment format;
- use of the report's basic concepts to develop a Site ALARA Enhancement Action Plan.

The beneficial effects of the improved work management approach induced by ISOE can also be seen in the continuous decrease of refuelling duration times. For example, US average refuelling duration was reduced from 55 days in 1990 to 32 days in 2000.

The importance of providing applied information in the native languages of the nuclear power plant personnel of different participating countries was recognised by the ISOE Steering Group. This report was therefore translated into several languages, including Chinese, German, Russian and Spanish, in addition to its standard version in English.

2.5 *Monetary value of collective dose*

During 1997, a survey was performed within the ISOE network to better understand the usefulness of the monetary value of collective dose in the practical application of protection optimization. This value is commonly referred to as the “alpha value”.

Eight regulatory authorities in charge of radiological protection (Canada, Czech Republic, Finland, Netherlands, Sweden, Switzerland, the United Kingdom and the United States) responded that they explicitly refer to the concept of monetary value of collective dose as a baseline reference for their regulatory decisions, and have defined one value or a set of values for this quantity. They also considered the implementation of the ALARA principle within the nuclear industry to be mainly an industry concern, and that, in this context, the monetary value of collective dose is essentially a managerial tool.

In most countries, alpha values are used when making decisions related to budget and impact on the operation and safety of a plant. About 60% of these uses are associated with significant modifications, large and expensive repairs, or chemistry of the plant.

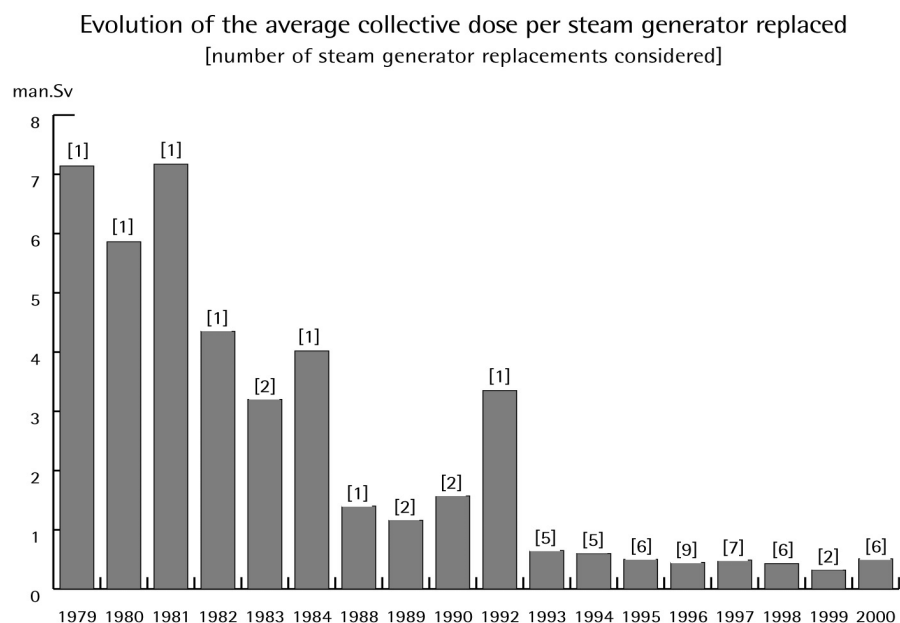
As of 1997, nearly three quarters of the operating organizations represented in ISOE had set up their own alpha-value system. Some use a single alpha value, the average of which is about US\$ 1 300 per man·mSv for North American utilities in the year 2000 and US\$ 600 per man·mSv for utilities in non-OECD countries. European utilities have established sets of monetary values which increase commensurate with increased risk. Mean values within this group, of about US\$ 1000 per man·mSv, do not differ drastically from those observed in the other groups.

Table 1. Alpha values used by utilities

	Alpha values US\$/[man·mSv]			
	Type	Minimum	Average	Maximum
North America (2000)	Single value	500	1300	3300
Europe (1997)	Set of values	17	1000	5300
Non-OECD (1997)	Single value	4	600	1000

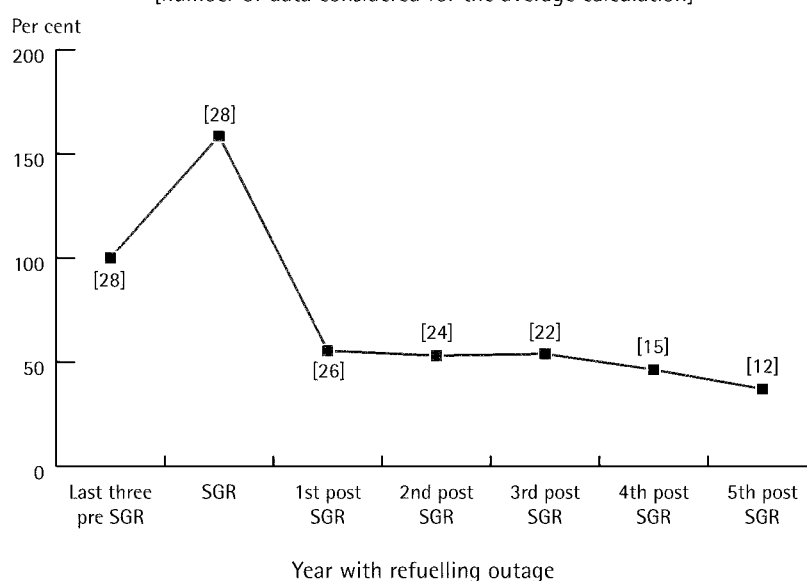
2.6 Steam generator replacements

Between 1979 and 2000, 58 steam generator replacements (SGR) were performed, mainly in North America and in Europe. Collective doses decreased regularly from more than 6 man·Sv per steam generator replaced in the late 1970s-early 1980s to an average of about 0.5 man·Sv during the last six years (see Fig. 3). However, that average masks quite large discrepancies and the best results correspond to three SGR performed in 1996 and 1998 in Belgium and France with only 0.21 man·Sv per steam generator replaced.



In order to evaluate the impact of a steam generator replacement on the evolution of the total annual collective dose for a reactor, the last three years with refuelling outages before each steam generator replacement were selected as a reference period. The average annual collective dose for each reactor considered over this reference period was normalised to 100. Collective doses for the steam generator replacement year and for the years with refuelling outages following the steam generator replacement were normalised accordingly (see Figure 4). The study showed that, on average, the collective dose during the year of steam generator replacement was 60% higher than the average collective dose during the three prior years with refuelling outages. The annual collective dose in the years following the SGR decreased to 40-60% of the pre-replacement three-year average collective dose.

Average impact of a SGR on the evolution of the reactor annual collective dose
[number of data considered for the average calculation]

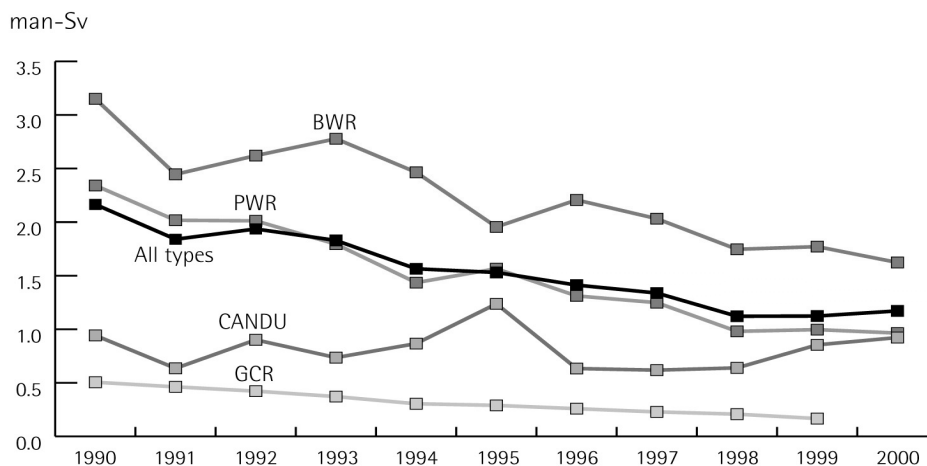


3. ISOE reveals downward dose trends

The annual average doses per reactor began to show a downward trend during the early years of nuclear power. Since the beginning of the ISOE Programme, this trend has been confirmed and consolidated, as can be seen in Fig. 5 showing data for the decade 1990-2000. Contributing to this trend are the improved communication and experience exchange between radiation protection managers of nuclear power plants worldwide, provided by the ISOE network, as well as the growing use of improved work management procedures developed and published through ISOE.

Although the data show some annual fluctuations, the average annual dose has been clearly decreasing for pressurised water reactors (PWR), from more than 2 man·Sv in the year 1990 to less than 1 man·Sv in 2000. For boiling water reactors (BWR), the dose came down from more than 3 man·Sv in 1990 to slightly over 1.5 man·Sv in 2000. The average annual dose for CANDUs in 1990 was already at a fairly low value of 1 man·Sv and has shown only some modest variations in the last decade. For gas-cooled reactors (GCR), the average annual collective dose, which was already lower than for other types of reactors, has continued to show a decreasing trend, from 0.5 man·Sv in 1990 to about 0.2 man Sv in 1999.

Average collective dose per reactor for operating reactors included in ISOE by reactor type for the years 1990-2000



The yearly fluctuations that can be seen in Figure 5 for all types of reactors are due to variations in outage scheduling, changes in cycle length and amount of maintenance work in the plants. For example, major work, such as the replacement of steam generators, leads to a significantly higher dose in the year of the replacement.

4. The future of ISOE

4.1 Improving the current system

During the last ten years, the ISOE Programme has gained a high level of participation and support. The major challenges the Programme still faces, in order to improve its current performance and effectiveness, are the need to complete the ISOE database as well as to further promote information exchange on actual examples, best practices and lessons learned in the field of occupational radiation exposure management.

As the ISOE database is the backbone of the Programme, it is essential for its success that the database is as comprehensive and updated as possible, containing detailed, up-to-date dose information for a variety of situations, jobs and tasks from all nuclear power plants worldwide. This completeness can be achieved only if and when all participants are motivated to input data that is as detailed as possible and to update their contributions regularly.

Information about experiences, lessons learned and best practices in occupational radiation exposure management for a large spectrum of situations should be shared amongst all participants as soon as the analysis of an interesting task is reasonably finalised. In order to facilitate this exchange of information and experience, important technical means have been developed to input relevant reports into the current database and, at the same time, to distribute the information through electronic media to all ISOE participants. Efforts have been made to achieve a system which is easy to use and not time-consuming. However, in the end, it is the commitment of participants to report on new

experiences and to share them with other radiation protection experts that determines the usefulness and success of the system.

Another important challenge here is the need to make sure that the two-tier information exchange scheme established by the Programme's Terms and Conditions can operate in a consistent and fair way. Careful management of the system is, in fact, necessary to ensure that the regulatory participants benefit from a fair share of information without, however, affecting the established right of the utility participants to preserve their own confidential channels for the direct exchange of detailed operational information.

4.2 *Addressing new challenges*

ISOE is also beginning to face new challenges where adjustments and expansion of the system may be required. These will have to address the increased importance of the decommissioning and dismantling of nuclear power reactors, as well as the discussion on future nuclear power plant generations. Plant life extension of currently licensed facilities will also be part of future concerns within ISOE. In all these areas ISOE can provide valuable information and a well- established community to discuss occupational exposure management issues.

As decommissioning and dismantling of nuclear power plants become more widespread, ISOE can play an important role in managing occupational radiation exposure during these activities. Information exchange on this growing issue and the use of analytical tools developed within ISOE will help achieve a higher level of protection for the workers involved in these activities. Information and experience contained within the ISOE system could also provide assistance in the design of new reactors, to ensure that an appropriate level of occupational dose management is built into their conception.

Another important concern for the future of ISOE is the establishment of liaisons with international organizations, such as the World Association of Nuclear Operators (WANO), to further improve the support of ISOE from nuclear power plant managers. Occupational radiation exposure in other areas of the nuclear fuel cycle – research reactors, fuel production, waste treatment – could be considered for future inclusion into ISOE.

5. Conclusions

During the first ten years, ISOE has gained a high level of participation, recognition and support. However, establishment of further interaction with international organizations, such as WANO, is needed with a view to further improving the support of ISOE from nuclear power plants. Active participation of a large number of operating organizations in this programme has contributed to a reduction in occupational exposure at nuclear power plants worldwide. In order to maintain or even further reduce the already low levels of occupational exposure, the ISOE system needs to be regularly used and further promoted and supported by its participants, both the utilities and the regulatory authorities. ISOE can also play an important role in achieving a high level of protection for the workers involved in decommissioning and dismantling of nuclear power plants.

References

- [1] OECD Nuclear Energy Agency, Occupational Exposure at Nuclear Power Plants, ISOE Annual Report, OECD, Paris (2001).
- [2] OECD Nuclear Energy Agency, ISOE – Information system on Occupational Exposure – Ten Years of Experience, OECD, Paris, (2002).
- [3] OECD Nuclear Energy Agency, Work Management in the Nuclear Power Industry: A Manual prepared for the NEA Committee on Radiation Protection and Public Health by the Information System on Occupational Exposure (ISOE) Expert Group on the Impact of Work Management on Occupational Exposure, OECD, Paris, (1997).

IMPLEMENTATION OF THE BASIC SAFETY STANDARDS IN THE REGULATIONS OF EUROPEAN COUNTRIES

C. Lefaure, P. Crouail

CEPN, Route du Panorama, BP 48, 92263 Fontenay-aux-Roses cedex, France

Introduction

The Euratom Basic Safety Standards for the radiological protection of workers and the general public against the dangers arising from exposure to ionizing radiation were laid down in Directive 96/29/Euratom adopted by the Council in May 1996. It should have been implemented in Member States before 13 May 2000. Other European countries should refer to the “International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources” issued in 1994 and jointly sponsored by FAO, IAEA, ILO, OECD/NEA, PAHO and WHO.

The objective of this information sheet, is to review the progress in implementing these Basic Safety Standards in the national regulations of European countries. This paper will describe specifically how the three fundamental principles of radiological protection have evolved (justification, optimisation and limitation).

The implementation of the European Directive was expected before mid-May 2000, most of the different Member States have today integrated it into their national laws. However, in those countries where it is not yet totally integrated, the projects are quite close to the final draft and will be therefore referred to in that presentation.

Table 1. Status of the Implementation of the Basic Safety Standards in the Regulations of European Countries (April 2002)

EC Countries	Progress in the implementation of the BSS	Date of implementation of the BSS
<i>Austria</i>	<i>Draft</i>	<i>Expected 2002</i>
Belgium	Implemented	20 July 2001
Denmark	Implemented	1 January 1998
Finland	Implemented	Before 13 May 2000
France	Partially Implemented	March 2001/April 2002
Germany	Implemented	1 August 2001
<i>Italy</i>	<i>Ready</i>	<i>1 January 2001</i>
Spain	Implemented	6 July 2001
Sweden	Implemented	1 December 2000
The Netherlands	Implemented	September 2001/ 19 February 2002
UK	Implemented	1 January 2000
NON EC COUNTRIES		
<i>Czech Republic</i>	<i>Ready</i>	<i>1 July 2002</i>
Hungary	Implemented	2000
Lithuania	Implemented	12 January 1999
Norway	Implemented	2000
Slovak republic	Partially Implemented	2001
<i>Slovenia</i>	<i>Ready</i>	<i>Expected 2002</i>
Switzerland	Implemented	1994
Ukraine	Implemented	1998

Justification principle

The justification principle is the first fundamental principle of the system of radiological protection recommended by the International Commission on Radiological Protection (ICRP). In the EURATOM Directive justification is not mentioned as a radiation protection principle, but as a “general principle”. It is the first “radiation protection requirement” in the International BSS.

Previous situation

In most regulations, this principle was not specifically addressed before the implementation of the new BSS. Instead, all practices actually implemented were implicitly considered as justified. However, some practices or trades were explicitly named as unjustified and consequently forbidden in the national regulatory texts. These included, for example, fluoroscopy for shoe-fitting, fishing floats, trade in beta lights (e.g. in the Netherlands), radioactive substances added in the production of foodstuffs, toys, personal ornaments and cosmetics (e.g. in Italy, Sweden, France) and lightning conductors (Italy, France). In Germany, there were no practices directly forbidden, however, there was

always agreement between the Federal Ministry and all the Länder Authorities on practices they would or would not authorize.

Implementation of the new BSS

Once the new BSS will be implemented, the justification principle will be explicitly stated in almost all-national regulations.

Wording

Member States shall ensure that all new classes or types of practice resulting in exposure to ionising radiation are justified in advance of being first adopted or first approved by their economic, social or other benefits in relation to the health detriment they may cause. Existing classes or types of practice may be reviewed as to justification whenever new and important evidence about their efficacy or consequences is acquired. (Council Directive 96/29/EURATOM, General Principles, Article 6.1 and 6.2)

No practice or source within a practice should be authorised unless the practice produce sufficient benefit to the exposed individuals or to society to offset the radiation harm that it might cause; that is: unless the practice is justified, taking into account social, economic and other relevant factors. (IAEA Safety series 115, International Basic Safety Standards, Principal Requirements § 2.20-§ 2.22)

A quick reading of the wording associated with the justification principle subsequently adopted in the European national regulations gives the impression that they are very close to the above. In fact, the wording used mostly reflects “cultural” differences.

In **Germany**, the justification principle was already stated in the former Radiation Protection Ordinance. However, it is now stated even more explicitly, closely following the wording in the European Directive.

In **Switzerland** (which does not belong to the EC), the justification principle is explicitly noted in the Federal Act on radiological protection (art. 8) and in the corresponding ordinance (art. 5).

In France, Denmark, the Netherlands and Norway (which does not belong to the EC), the justification principle is applied to a very large set of human activities (and goes beyond articles 6.1 and 6.2 of the European Directive):

“The economic, health or other benefits that arise from an activity or an intervention shall be greater than their inherent inconveniences”(France, Sweden).

“The benefit should outweigh the health damage. If not justified, a practice is not allowed.”(The Netherlands).

“At every use of radiation the advantages shall go beyond the risks”(Denmark).

“Any human activity involving radiation sources has to be defensible: the benefits of the activity shall exceed the risks associated with the radiation”(Norway).

In Finland and Sweden, the justification principle applies mainly to practices (a practice is a human activity that can increase the exposure of individuals to radiation):

“The benefits accruing from the practice shall exceed the detriment it causes”(Finland).

“Anyone who conducts a practice with ionising radiation shall ensure that the practice is justified by which is meant that the use of radiation gives a benefit that exceeds the estimated health detriment caused by the radiation”(Sweden).

In Spain, Belgium and Slovenia the justification principle is mentioned for new practices:

“All new classes or types of practice involving exposure shall be justified by the promoter to the competent Authority, which will then decide on [...] its adoption considering the benefits in relation to the health detriment they may cause”(Spain).

“The different types of practices leading to ionising radiation exposures shall be justified before the first adoption or the first authorisation, taking into account and balancing the corresponding advantages and drawbacks, including the health aspects”(Belgium).

In Ukraine, the wording concerning justification includes within the evaluation of the harm the occurrence of a critical event (accident) and the willingness to take care of the future:

“a practice which can lead to exposure to ionising radiation shall not be implemented if the benefit for the people exposed and society in general does not exceed the harm from this activity now and in the future in connection with the potential occurrence of critical event”

Austria is the only country where it is stated that established practices are considered justified as long as no important new insights prompt reconsideration. Application of new practices has to be justified.

In the United Kingdom, the justification principle has not previously been explicitly addressed in occupational exposure legislation. It is recognised that an appropriate legal instrument will have to address this. However giving the justification principle legal force within the UK legislative system has posed a number of regulatory enforcement issues. A proposed way forward is currently being considered by Ministers.

Legal requirements

Some national Authorities have specified regulatory requirements for enforcing the justification principle: these include lists of justified and unjustified practices, evaluation procedures of practices, etc.

In Germany, some practices (for example, the irradiation of filters from water supply stations with ⁶⁰Co sources which was a common practice in East Germany before the reunification) or particular uses of radiation (consumer products such as ordinary watches containing radioactive material) will be explicitly forbidden in the “administrative provisions” which accompany the implementation of the rules laid down in the Ordinance. The decision whether a practice is justified or not is taken by the Federal Ministry of Environment, Nature Conservation and Nuclear Safety on the basis of a common understanding with the Länder Authorities.

In **Belgium**, before the acceptance of a new activity or practice, it is now mandatory to undertake a justification study that can be reviewed by the competent authority.

In **France**, it is now clearly stated that the competent authority in pursuance of the justification principle could forbid a nuclear activity.

In **Spain and Slovenia**, the authority may propose to review the justification of existing practices whenever new and important evidence about their efficiency or consequences is revealed. In Spain the justification of a new practice has to be approved by the competent Authority, e.g. the Government Departments and by the CSN. The CSN is the only competent Authority for the justification revision of existing practices.

In the **Netherlands**, there will be a ministerial Ordinance with a list of justified and a list of non-justified practices and work activities. If the activity is not on the list as a “justified practice”, it will be forbidden, unless a request for justification, with good supporting arguments, is approved.

In **Switzerland**, activities involving ionising radiation leading to an effective dose less than 10 $\mu\text{Sv}/\text{year}$ shall always be regarded as justified.

The justification principle is now re-emphasised in nearly all countries regulations. This is accompanied by a stronger control by Authorities of activities involving radioactive substances.

Optimization principle (ALARA)

The optimization principle has been reemphasized as the core of the system of radiological protection in the ICRP Publication 60 and in the European Basic Safety Standards.

Previous Situation

The optimization principle was already stated in most national laws, albeit in general terms, often without any practical guidance (but in countries like the UK through an approved code of practices). Consequently, the application of optimization for practices was often quite limited.

Implementation

The implementation of the new BSS appears to provide both the Authorities and users of ionizing radiation sources with more precise guidance on how to apply the optimization principle.

Wording

“In the context of optimization [Member States shall ensure that] all exposures shall be kept as low as reasonably achievable, economic and social factors being taken into account”. (Council Directive 96/29/EURATOM, General Principles, Article 6.3)

“In relation to exposures from any particular source within a practice, except for therapeutic medical exposures, protection and safety shall be optimised in order that the magnitude of individual doses, the number of people exposed and the likelihood of incurring exposures all be kept as low as reasonably achievable [ALARA], economic and social factors being taken into account, within the restriction that the doses to individuals delivered by the source be subject to dose constraints”. (IAEA Safety Series 115, International Basic Safety Standards, Principal requirements § 2.24)

In the **Netherlands**, “the undertaking shall ensure that the equivalent or effective dose to individuals, taking account of the number of exposed individuals, due to a practice is as low as reasonably achievable. The undertaking shall ensure that, regarding the potential exposures, both the doses in the case of an exposure and the probability of an exposure is as low as reasonably achievable. With regards to this Decree and all related requirements, for the assessment of what is ‘reasonably achievable’, economical and social aspects shall be taken into account.”

In the **United Kingdom**, “every radiation employer shall, in relation to-any-work with ionising radiation’s that he undertakes, take all necessary steps to restrict so far as is reasonably practicable, the extent to which his employees and other persons are exposed to ionising radiation”.

This wording is unchanged from the previous regulations.

In **Spain**, “the magnitude of individual doses, the number of people exposed and the likelihood of incurring exposures, shall be kept as low as reasonably achievable, economic and social factors being taken into account.”

In **Finland**, “the practice shall be organised in such a way that the resulting exposure to radiation hazardous to health is kept as low as reasonably achievable.”

In **Denmark**, “all doses shall be as low as reasonably achievable.”

In **Belgium**, “all exposures shall be kept as low as reasonably achievable, taking into account social and economic factors”.

In **France**, “exposure of individuals to ionising radiation’s shall be kept as low as reasonably possible, according to -the technical state of the art, - economic and social factors -and eventually medical goals”(Ordinance March 2001) “

In **Sweden**, “anyone who conducts a practice with ionising radiation shall ensure that the radiation protection measures are optimised, which means that exposures of people are as low as reasonably achievable, economic and social factors being taken into account.”

In **Italy**, there is no new wording of the ALARA principle: the ALARA principle was already mentioned with reference to exposures of workers and persons of the public and to technical requirements the installations must fulfil.

In **Germany**, in the new Ordinance, the ALARA principle is stated unchanged and as a general guidance, which is, however, legally binding in all cases. The wording is: “... also below the dose limits, unnecessary radiation exposure or contamination of men and environment should be kept as low as possible, according to the latest technical and scientific standards and taking into consideration all conditions related to an individual case.” In fact, German law promotes the

“minimisation” principle together with the “principle of proportionality”, which means: doses are reduced to levels as low as reasonably possible.

In **Norway**, the basic principles, justification, optimisation and dose limitation, are stated in a general article with a requirement that any human activity involving radiation sources has to be defensible. It is stipulated that the activity must be prepared to avoid acute effects and to minimise the risks for late injury as low as reasonably achievable.

In **Switzerland**, the conditions for realising the optimisation principle are described in the Radiological Protection Ordinance (art. 6).

In **Ukraine**, “critical event probability and potential exposure as well as the number of persons that could be impacted by ...sources shall be as low as reasonably achievable taking into account economic and societal considerations”

Although in many cases the evolution of the optimisation principle wording is not revolutionary, it refers now explicitly to economic and social factors in many countries and as well mentions explicitly in a few cases patient exposure.

Guidance for practical applications

In addition to the basic regulatory requirement that exposures have to be optimised, regulators have increasingly introduced guidance on how this principle should be applied in practice.

For example, in **France**, a specific Decree concerning the protection of workers against ionising radiation (Decree n° 98-1185 modifying the Decree n° 75-306, Art. 20 bis) says, that in order to implement ALARA: “work stations which expose workers to ionizing radiation’s shall be analyzed periodically to review the doses received. The frequency of these reviews must be a function of the level of the doses. In particular, during an operation in a controlled area, the manager of the plant in collaboration with the employer – if he is not the manager – is in charge of:

- a prior assessment of the collective and individual doses that might be received by workers,
- having the actual doses received during the operation registered and analyzed in order to draw conclusions from the radiation protection point of view; if it is technically possible, these measurements should be made in real time with immediate reading devices (“the operational dosimetry”).

For the prior assessment of doses, the draft of another Decree specifies that “the radiation protection qualified expert in conjunction with the persons responsible for the operation, shall define individual and collective doses targets (which are not comparable to the regulatory limits)”.

In the **Netherlands**, a dose prediction has to be performed by undertakings when requesting a licence and when planning work activities, with regards to members of the public off site and to workers on site. Authorities evaluate these predictions and sometimes more reduction is required. Most sites are required to give a yearly overview of the real time measures or calculations both for workers on site and for members of the public off-site.

In the **Swedish** regulations it is stipulated that, in order to demonstrate the compliance with the optimisation principle, the licence-holder shall ensure that appropriate goals and control actions are

established and documented and that the necessary resources are available (*SSI Code of Statutes, SSI FS 2000:10, Regulations on Radiation Protection of People Exposed to Ionising Radiation at Nuclear Plants*). The goals and control actions shall be appropriate to the particular plant and be drawn up to take care of daily as well as long-term radiation protection. All individuals that are exposed to ionising radiation or are decision makers in matters that affect the individual doses shall be informed of the goals and the means of control. The practice, including the goals and control actions, shall regularly be followed up and evaluated. Such evaluations shall be performed at least once a year. Documentation on the evaluation shall be sent to the Swedish Radiation Protection Institute.

In **Finland**, the radiation exposure to which workers are subjected and the factors affecting it, shall be assessed in advance, also taking into account exceptional working conditions.

In **Spain**, CSN has approved a new guide within the Nuclear Power Plants Safety Series where the main recommendations regarding the management of radiation exposure optimisation are presented. This guide comprises the ALARA responsibility assignments to all the involved parties. Besides a well established ALARA policy, it is necessary to implement a set of actions, called ALARA program, to be addressed by the licensee such as ALARA goals, work management, source term control and reduction, ALARA review of design modifications, special training and internal audits. The guide covers these aspects in a wide and flexible way to be adaptable to different circumstances. This document applies to utilities and contractors involved in all the phases of activity in nuclear power plants: design, construction, operation, dismantling and modifications.

In **Lithuania**, the Hygiene Standard HN 87 2001 requires the establishment and implementation of an ALARA programme with:

- proper work organisation;
- improvement of working conditions;
- perfection of technological processes;
- training of personnel;
- implementation of quality insurance programme;
- improvement of safety culture;
- evaluation of influence of “human factor”.

In **Slovenia**, the future law points out that “prior evaluation of the risk and optimisation of radiological protection” should be performed in all working conditions.

In the **UK**, IRR99 are supported by an Approved Code of Practice (ACoP), which has a legal significance and by Guidance material, that though having no legal significance gives a very strong indication of what is practically needed to demonstrate compliance. Prior risk assessment is mandatory in the UK: “Before a radiation employer commences a new activity involving work with ionising radiation ... he shall make a suitable and sufficient assessment of the risk to any employee and other persons for the purpose of identifying the measures he needs to take to restrict the exposure of that employee or other person to ionising radiation. [...] A radiation employer shall not carry out work with ionising radiation unless he has made an assessment sufficient to demonstrate that all hazards with ionising radiation have been identified; and the nature and magnitude of the risks to employees and other persons arising from those hazards have been evaluated”. The ACoP specifically requires, where relevant, the risk assessment to include several factors including “the estimated dose rates to which anyone can be exposed” and to take into account “the results of any previous personal dosimetry or area monitoring relevant to the proposed work”.

In **Germany**, the Ordinance was already supported by guidelines issued by the Federal Minister of Environment. For example, the guidelines on radiation protection of maintenance and repair of work in light water reactors gives guidance on what is necessary in order to minimise doses. The estimated collective dose for each Nuclear Power Plant for the following year is required for plant personnel and contractors. If predicted collective doses are higher than 50 man·mSv, or individual doses higher than 10 mSv, specific procedures are required (job planning, step-by-step time and dose calculation, discussion with authority experts, preparation of protection actions, close supervision during the work, stopping the work and new planning if problems occur, step-by-step documentation on job time, dose values and radiological measurements).

The optimisation principle has grown into a stricter regulatory requirement in almost all new regulations, including prior dose assessment, operational dosimetry, information of stake holders, ALARA responsibility assignments.

Limitation

Dose limits for deterministic Effects

There are no major changes to the limits for avoiding deterministic effects. For workers, the limit in terms of dose equivalent to the lens of the eye is 150 mSv/year (50 mSv/year for minors). In terms of dose equivalent to the skin the limit is 500 mSv/year (generally over 1 cm² of skin instead of 100 cm² in the past; 150 mSv/year for under age people); and in terms of dose equivalent to the hands, forearms, feet and ankles, the limit is 500 mSv/year (150 mSv/year for under age people). In Germany, there are also organ dose limits for gonads, uterus and red bone marrow (50 mSv/year); thyroid and bone surface (300 mSv/year); colon, lung, stomach, bladder, breast, liver, oesophagus and other organs and tissues (150 mSv/year). In Germany, in specific circumstances the limit is 300 mSv for the lens of the eye, and 1000 mSv for other organs.

Dose limits for stochastic effects

Table 2 gives the new individual dose limits in the countries that have already implemented the BSS, and the most recent drafted values in the other European countries that have yet to implement them.

All countries have, or will have, a dose limit for the public that is 1 mSv per year, Denmark and Finland specifying that such a limit corresponds to the contributions of all sources together. However, some countries have been or will be more restrictive with regards to each source. The Lithuania, UK, Germany and the Netherlands have introduced some constraints and specified that each source may not contribute to more than 0.2, 0.3, 0.3, and 0.1 mSv per year.

The situation is somehow different in the case of occupational exposure limits. The interpretation of the BSS has led the countries to select either 100 mSv for five years with a maximum of 50 mSv per single year (Finland, Spain, Sweden, Czech Republic, Switzerland), or to be more stringent in selecting 20 mSv per calendar year (Denmark, Germany, Italy, the Netherlands, UK, Norway) or per 12 consecutive months (Austria, Belgium, France).

One country has introduced an annual averaged dose limit of **10 mSv**:

400 mSv over the work life in Germany

Table 2. Dose Limits for Stochastic Effects (mSv)

Countries	Members of Public	“Workers A” and Major Students	“Workers B” and Minor Students	Pregnant Women and Foetus	Workers in exceptional circumstances (excluding emergency situations)
EC EURATOM DIRECTIVE 96/29	1 / year	100 / 5 years & 50 / year	6 / year	1 (foetus)	–
<i>Austria</i>	<i>NA</i>	<i>NA</i>	<i>NA</i>	<i>NA</i>	
Belgium	1 / year	20 / 12 rolling months	6 / year	1 (foetus) and if likely >1 women work outside controlled areas	2 x annual limits per operation / 12 rolling months & < 5 x annual limits (doses already received included)
Denmark	1 / year 0.1 / source	20 / year	6 / year	1 (foetus)	–
Finland	1 / year	100 / 5 years & 50 / year	6 / year	1 (foetus)	–
<i>France</i>	1 / year	20 / 12 rolling months	6 / year	1 (foetus)	2 x annual limits per operation
Germany	1 / year 0.3 / site	20 / year 400 / lifetime	6 / year	1 (foetus), 2/month (uterus)	-100 per year
Italy	1 / year	20 / year	6 / year	?	?
The Netherlands	1 / year 0.1 / source	20 / year	6 / year	unlikely > 1 (woman) **	100 / operation
Spain	1 / year 5 / 5 years *	100 / 5 years & 50 / year	6 / year	1 (foetus) & unlikely > 1 (woman) **	case by case (needs CSN approval)
Sweden	1 / year	100 / 5 years & 50 / year	6 / year	1 (foetus) **	case by case (needs SSI approval)
UK	1 / year 0.3 / source	20 / year	6 / year	1 (foetus) 13 / 3 months (abdomen equiv. dose) ***	100 / 5 years & 50 / year
INTERNATIONAL BSS (1994)	1 / year	100 / 5 years & 50 / year	6 / year	-	200/10 years & 50/year (review when over 100) or 50/year renewable 5 times
<i>Czech Rep.</i>	1 / year 5/5 years *	100 / 5 years & 50 / year	6 / year	1 (foetus) unlikely > 1 (woman) **	50 / year (“specific circumstances”) 500/5 years (“unusual events”)
Hungary	1/year	100 / 5 years			
Lithuania	1 / year 5/5 years *	100 / 5 years & 50 / year			
Norway	1 / year	20 / year	6 / year	?	?
Slovak Rep.		100 / 5 years & 50 / year			
<i>Slovenia</i>	1 / year	NA			
Switzerland	1 / year	100 / 5 years & 50 / year	5 / year	2 (abdomen surface effective dose)	100 / 5 years & 50 / year
Ukraine	1 / year	20 / Year **** 100 / 5 years			

Italic characters: not yet implemented.

* in specific cases.

** for the remainder pregnancy period.

*** for women of reproductive capacity.

**** 20 for new facilities; 50 for operating facilities with transition to 20.

Conclusion

The full implementation of the BSS across Europe into national regulations is not far to be achieved. In addition, the principles of justification, optimisation and dose limitation have to be incorporated into a number of very different national regulatory structures. Despite this, there is evidence to suggest that all three principles will be applied across Europe in a much more consistent manner than previously, as a result of the new BSS.

Justification is probably the biggest change since it was commonly excluded from previous regulations. The optimisation principle has been translated into the different national structures in a consistent manner.

More significantly, there is increasing emphasis on applying and demonstrating optimisation in practice, in either the regulations or supporting guidance.

The flexibility in the BSS for setting effective dose limits has been reflected in national regulations. Consequently, different European countries specify either a 1 year or a 5 years effective dose limit, or a combination of both. In practice, where the optimisation principle is observed, these differences are not expected to cause practical difficulties.

The implementation of the BSS into practice appears now to be the on going challenge.

Annex

Regulatory references in European countries

Countries	Draft legislation	Regulatory References
Austria		Strahlenschutzgesetz, BGBl. Nr. 227/1969, last modified BGBl. Nr. 16/2000.
Belgium		Arrêté royal portant mise en vigueur de la loi du 15 04 1994 relative à la protection de la population et de l'environnement contre les dangers résultant des rayonnements ionisants et relative à l'Agence fédérale de contrôle nucléaire. 20 July 2001
Denmark	–	National Board of Health, Order no. 823 of 31 October 1997 on dose limits for ionising radiation
Finland	–	- Revised Radiation Act (1142/1998) - Revised Radiation Decree (1143/1998) + "Radiation Safety in Practices Causing Exposure to Natural Radiation" (STUK Guide, April 2000)
France	+ 2 other Decrees for the workers and the patient	- Ordinance 2001 270 28 March 2001 "relative à la transposition de directives communautaires dans le domaine de la protection contre les rayonnements ionisants" - Decree 2002 460 from the 2002.04 04 "protection générale des personnes contre les dangers des rayonnements ionisants".
Germany	Strahlenschutzverordnung + Codes of Practice	Verordnung für die Umsetzung von EURATOM-Richtlinien zum Strahlenschutz (20 July 2001)
Italy		Decreto Legislativo no 241/2000 (27 May 2000)
the Netherlands		Radiation Protection Decree (Besluit Stralingsbescherming), State Journal (Staatsblad) 2001, nr 397 of 06-09-2001. It came into force on the first of March 2002, Staatsblad 2002, nr 81 of 19-02-2002.
Spain		- Regulation of nuclear and radioactive facilities (Royal Decreto 1836/1999, 3 December 1999) - Regulation for the protection of health against ionising radiations (Royal Decreto) (20 July 2001)
Sweden		Revised Radiation Protection Act (SFS 1988:220) 13 May 2000 - Revised Radiation Protection Ordinance (Swedish Code of Statutes SFS 1988:293) + Regulations of the Swedish Radiation Protection Institute (SSI) for implementing 96/29 (29 October 1998): SSI FS 1998:3 (Categorisation of workplaces and workers at work with ionising radiation), SSI FS 1998:4 (Dose limits at work with ionising radiation), SSI FS 1998:5 (Monitoring and reporting of individual radiation doses), SSI FS 1998:6 (Medical examination for work involving ionising radiation)
United Kingdom	–	- Ionising Radiation Regulations 1999 (replaces IRR85) + Approved Code of Practices (ACoP)
Czech Republic	amendment	Act No. 18 / 1997 Coll. on Peaceful Utilisation of Nuclear Energy and Ionising Radiation (the Atomic Act) and on Amendments and Additions to Related Acts. Regulation No. 184 / 1997 Sb. of the State Office for Nuclear Safety on Radiation Protection Requirements
Hungary		16/2000 EüM rendelet/Order of HealthMinistry No. 16/2000
Lithuania		Law on radiological protection (N°VIII-1019, 1999) Hygiene Standards HN 73:2001 "Basic Standards of Radiation Protection"; 2001
Norway	–	? (based on ICRP 60 and IAEA BSS 115) since 1 July 2000
Slovenia	New Law on radiation and Nuclear Safety	
Slovak Republic		Radiation Protection December 2001, Ministry of Health
Switzerland	–	- Federal Act on Radiological Protection (March 1991), - Ordinance on Radiological Protection (22 June 1994)
Ukraine		"Protection of man from acting ionising radiation about" law 24 February 1998 "Radiation safety standard of Ukraine" RSSU 97 01 01 1998

IMPLEMENTATION OF THE INTERNATIONAL BASIC SAFETY STANDARDS IN IAEA MEMBER STATES: A UNIQUE EXPERIENCE

M. Gustafsson, K. Mrabit
International Atomic Energy Agency, Austria

1. Abstract

During the last seven years, an unprecedented proactive international co-operative effort has been implemented in 52 Member States of the International Atomic Energy Agency (IAEA) within the framework of its Technical Co-operation (TC) Model Project on Upgrading Radiation Protection Infrastructure. The objectives of this Model Project are to assist those Member States which have an inadequate radiation and waste safety infrastructure and are already receiving IAEA assistance, in complying with the International Basic Safety Standards for Protection against Ionising Radiation and for the Safety of Radiation Sources. These objectives are in line with the statutory mandate of the IAEA, which stipulates that safety standards are to be applied to its own operations, including all TC activities.

Five project milestones were defined in order to facilitate the setting of priorities, the timing and monitoring of progress, the optimisation of resources, and to achieve compliance with the BSS. These milestones comprise the establishment of (1) a regulatory framework; (2) occupational exposure control; (3) medical exposure control; (4) public exposure control; and (5) emergency preparedness and response.

By the end of September 2001, more than 75% of participating Member States had promulgated radiation protection laws and established regulatory authorities, and over 60% had put in place individual and workplace monitoring control. During the last seven years of implementation, more than 900 fellows and scientific visitors received individually tailored training; 3 000 persons attended educational and specialised courses; 1 150 expert and monitoring missions were undertaken; and equipment worth US\$7 millions was provided.

The purpose of this paper is to present the approach used and the current status of this Model Project.

2. Background

By its Statute, the IAEA is authorised to establish or adopt safety standards for protection of health and minimisation of danger to life and property, and to provide for the application of these standards to its own operations as well as to operations making use of materials, services, equipment, facilities, and information made available by the IAEA.

The safety standards of reference are the *International Basic Safety Standards for Protection against Ionising Radiation and for the Safety of Radiation Sources* (BSS), the latest revision of which was published in 1996 [1]. The application of the standards is done through, inter alia, provision of technical assistance within the framework of the IAEA (Agency) Technical Co-operation (TC) programme. The IAEA's Statute further requires that its Board of Governors consider the "adequacy of proposed health and safety standards for handling and storing materials and for operating facilities" before giving approval to TC projects.

For more than a decade, from 1984 to 1995, information specifically relevant to radiation safety was obtained through more than 60 expert team missions undertaken by the Agency's Radiation Protection Advisory Teams (RAPATs) and follow-up technical visits and individual expert missions. The RAPAT programme documented major weaknesses and the reports provided useful background for the preparation of national requests for IAEA technical assistance.

Building on this experience and subsequent policy reviews, the IAEA took steps to evaluate more systematically the needs for technical assistance in areas of radiation and waste safety. The outcome was the development of an integrated system designed to more closely assess national priorities and needs for upgrading radiation and waste safety infrastructures.

This paper reviews the IAEA's integrated approach used and the status of the *Model Project on upgrading radiation protection infrastructures* (hereafter called the Model Project) in its Member States. The project is being implemented under the TC programme and involved 52 countries in its first phase.

3. The model project

3.1 Project objectives

The objectives of the Model Project are to assist those IAEA Member States, which have an inadequate radiation and waste safety infrastructure and are already receiving IAEA assistance, so that they can comply with the BSS.

The project was first approved in 1994 as an interregional project with the involvement of five Member States. It was enlarged in 1995 to include 52 Member States that were grouped into five regional projects and managed by four Regional Project Managers [(17 in Africa; 11 in Europe; 10 in Latin America; 14 in Asia (5 in East Asia and 9 in West Asia)]. One of the first actions in implementing the project was to define more clearly what constituted an adequate radiation and waste safety infrastructure. This had to be done for different types of radiation applications ranging from common industrial and medical uses and practices found in every country to the full nuclear fuel cycle which exists in relatively few developing countries. Decisions were taken about what was needed to bring each country up to an adequate level, about how to implement the provision of technical assistance and how to verify results.

The main components of this process consist of collecting and evaluating information on the existing safety infrastructure, establishing and maintaining Country Radiation and Waste Safety Profiles and formulating and implementing Country Safety Action Plans. For all participating countries, assessments were made to identify their infrastructure weaknesses. These included establishment of (1) a regulatory framework; (2) occupational exposure control; (3) medical exposure control; (4) public exposure control; and (5) emergency preparedness and response, and are later

referred to as the five project milestones. The Regional Project Managers discussed shortcomings with national authorities as part of the steps to prepare and finalise detailed Country Safety Action Plans (needed to rectify weak or non-existent infrastructure elements, to monitor the development of improvements in safety infrastructure, to sustain an effective infrastructure and develop it for additional uses of radiation). In each Action Plan, the obligations both of the participating country and of the Agency were specified for the effective implementation of the projects with the planned time frame.

Countries participating in the model projects to upgrade radiation and waste safety infrastructure				
Africa	East Asia & the Pacific	West Asia	Europe	Latin America
Cameroon	Bangladesh	Kazakhstan	Albania	Bolivia
Cote d'Ivoire	Mongolia	Jordan	Armenia	Costa Rica
Ethiopia	Myanmar	Lebanon	Belarus	Dominican Republic
Gabon	Sri Lanka	Qatar	Bosnia & Herzegovina	El Salvador
Ghana	Viet Nam	Syrian Arab Republic	Cyprus	Guatemala
Madagascar		Union Arab Emirates	Estonia	Haiti
Mali		Uzbekistan	Georgia	Jamaica
Mauritius		Yemen	Latvia	Nicaragua
Namibia		Kyrgyzstan	Lithuania	Panama
Niger			Moldova	Paraguay
Nigeria			Former Republic of Macedonia	
Senegal				
Sierra Leone				
Sudan				
Uganda				
Dem. Rep. of Congo				
Zimbabwe				

3.2 Country radiation and waste safety profiles

The intention behind the establishment of a Country Radiation and Waste Safety Profile information system is to maintain and keep updated all the data and information known to the Agency on the radiation and waste safety infrastructure of a given Member State receiving its assistance. The essential structure of the system relies on a questionnaire, the answers to which are the basic inputs for the computerised database.

The questionnaire and derived database cover the following main sections:

- organisational infrastructure;
- legal and regulatory status;
- extent of practices involving ionising radiation;
- provisions for individual dosimetry;
- public exposure control;
- radiation protection and safety of patients in medical diagnosis and therapy;
- transport of radioactive material;
- planning and preparedness for radiation emergencies;
- quality assurance; and
- education and training.

However, the system is not limited to the database, but also includes a narrative of the status of radiation and waste safety infrastructure, the assembly of hard copies of information including laws and regulations, mission reports, and other material, as well as relevant Country Safety Action Plans.

4. Methodology of implementation

4.1 Commitment by the governments

It should be noted that the Model Project presumes that governments and national authorities are prepared to comply with their obligations as described in the Preamble of the BSS. For this reason, Country Safety Action Plans were approved by relevant counterparts and authorities in each participating Member State and the implementation of the Country Safety Action Plans did not start before official approval from the Member State concerned was obtained. As a result of this approach, Member States firmly committed themselves to establishing a national infrastructure, which includes *inter alia*:

- appropriate national legislation and/or regulations (the type of regulatory system will depend on the size, complexity and safety implications of the regulated practices and sources as well as on the regulatory traditions in the country);
- a regulatory body empowered and authorised to inspect radiation users and to enforce the legislation and/or regulations;

- sufficient resources;
- adequate numbers of trained persons; and
- an adequate occupational radiation protection programme.

4.2 Country safety action plans

The Country Safety Action Plans were thus developed from an analysis of the Country Radiation and Waste Safety Profiles against the requirements for an adequate safety infrastructure. Missing or deficient items were determined and documented for the preparation of a Safety Action Plan specific to each country, and including actions that are needed for the country to achieve a full and adequate infrastructure commensurate with its existing and planned applications of ionising radiation.

The Action Plans include both generic and specific activities. Generic activities apply to all countries and as a first priority cover notification, authorisation, and control of all radiation sources – whatever their use – within the country. Later steps cover protection of workers, patients receiving medical treatment and the public from environmental releases; emergency plans; transport arrangements; and other areas. Specific activities are tailored to each country's particular needs, such as personnel training or the provision of necessary equipment.

The development of human resources through training is an important component of the Model Project. It involves not only educational courses but covers also specialised training for, *inter alia*, regulators, radiation protection specialists, and medical personnel. The establishment and sustainability of a sound infrastructure for assuring radiation and waste safety depends heavily upon national capabilities in education and training in these areas.

4.3 Generalising the system

The first milestone to be achieved under the Model Project is the establishment of a system of notification and authorisation as required by the BSS, followed by milestone 2 on occupational exposure control. The Regional Project Managers are expected to monitor and report on each country's compliance with the respective milestones, and the Agency reports regularly to its Board of Governors on the progress achieved; the latest report was submitted in November 2001.

The approach used for the implementation of the Model Project represents a system that is being generalised to all Member States receiving IAEA assistance. It will provide the Agency with a fully documented system for assessing the current status of any country with respect to its radiation and waste safety infrastructure and a prioritised and agreed set of needs that should form the basis of future technical assistance activities. There will also be enough data to assess the capacity of the country to assure the safety of other developments of technology or requested items of equipment that could pose radiation hazards.

Over time, the system should provide a firmer basis for the Agency's co-operative work with its Member States and the provision of technical assistance in areas of radiation and waste safety. Efforts can be better directed towards achieving a situation in which no Member State which actively co-operates with the IAEA can have an inadequate radiation and waste safety infrastructure. Under an agreed action plan, this work will encompass measures for improving the identification of needs and

requirements and enhancing the use of resources to further strengthen national capabilities for ensuring safety in the peaceful applications of nuclear and radiation technologies.

4.4 *Standardisation of activities*

In order to manage this big undertaking, the Agency must make efficient use of resources, which implies a balance between standardised measures and respect for the peculiarities of each Member State. As described herein, a number of activities have been standardised. This includes, inter alia:

1. The Country Safety Action Plans contain the same elements for all Member States although individual actions may differ depending on the country profiles used for tailoring the action plans.
2. Model legislation and model regulations ensure a consistent and coherent international approach, and yet national legal traditions are respected by allowing for local adaptation.
3. Checklists for the safety review of the main practices using radiation have been provided simultaneously to more than 50 Member States.
4. Training of manpower for the users and regulators is being done in a synchronised and standardised manner, through regional educational and specialised training events, and information exchange is being fostered through regional workshops and seminars.
5. The setting of milestones facilitates a common methodology and timing to monitor progress, as described below.
6. An information system for regulatory authorities is being implemented simultaneously in more than 50 Member States. This system is called Regulatory Authority Information System (RAIS).

5. Monitoring progress

5.1 *Peer reviews and appraisal services*

As the implementation of the Country Safety Action Plans progresses, both Member States and the IAEA need to appraise in a systematic and harmonised way the effectiveness of the measures taken in order to correct weaknesses and optimise resources. For this purpose, and in addition to the regular monitoring made by the Regional Project Managers, the Agency established:

- an IAEA-TECDOC-1217, *Assessment by Peer Review of the Effectiveness of Regulatory Programme for Radiation Safety* [2], which provides advice on the conduct of peer reviews using a methodology to obtain qualitative and quantitative information and on its analysis against performance criteria and indicators; and
- an Occupational Radiation Protection Appraisal Service (ORPAS) that aims at assessing the regulatory and practical implementation of occupational radiation protection arrangements, and also at identifying specific strengths and best practices that can be shared with other Member States. It also provides a basis for determining where improvements may be required and for recommending actions to make such improvements.

6. Results achieved up to September 2001

From 1999 until September 2001, 32 Peer Review missions were undertaken that provided an independent assessment of the project achievements. The review teams' most relevant findings were in line with the assessment done through the continuous project monitoring made by the Agency.

It could thus be concluded that substantial progress has been made in upgrading radiation protection infrastructure, especially in the regulatory framework and occupational exposure control. By the end of September 2001, more than 75% of participating Member States had promulgated radiation protection laws and established regulatory authorities, and over 60% had put in place individual and workplace monitoring control.

With respect to the results achieved, the participating countries may be divided into three categories:

- Countries advanced in project implementation, which have attained milestones 1 and 2 and have succeeded in implementing several activities related to the other milestones.
- Countries where there have been some implementation delay; these countries need to revise existing legislation and restructure existing radiation protection systems including occupational radiation protection. There are indications that the national authorities concerned have become more committed, and that steps have been taken to speed up implementation. It is expected that, if this trend continues and there is no serious delay, these countries should be able to report substantial progress in meeting the principal requirements of the BSS in the foreseeable future.
- Countries where there have been major implementation delays as a result of institutional instability, severe general infrastructural weaknesses, inadequate support at the decision-making level, changes in national programme priorities, etc. These countries had not established a national legislative and regulatory infrastructure to be able to regulate practices and enforce national legislation and regulation that govern, among other things, occupational radiation protection activities.

It is worth mentioning that by the end of September 2001, the Agency has received requests from 29 Member States (additional to the 52 Member States which participated in the first phase of the Model Project), thus the total number of Member States participating in the Model Project now exceeds 80. The Agency, as requested by its Board of Governors, will integrate any new Member State, upon request, into relevant regional Model Projects on upgrading radiation protection infrastructure.

Reference

- [1] International Atomic Energy Agency, Food and Agriculture Organization of the United Nations, International Labour Organisation, OECD Nuclear Energy Agency, Pan American Health Organization, World Health Organization, "International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources", Safety Series No. 115, IAEA, Vienna (1996).
- [2] International Atomic Energy Agency, Assessment by Peer Review of the Effectiveness of Regulatory Programme for Radiation Safety, IAEA TECDOC Series No. 1217, Vienna (2001).

THE NEW GERMAN RADIATION PROTECTION ORDINANCE

W. Pfeffer

Gesellschaft für Anlagen- und Reaktorsicherheit

G. Weimer

Geschäftsstelle der Reaktor-Sicherheitskommission beim Bundesministerium für Umwelt,
Naturschutz und Reaktorsicherheit

Introduction

According to European law, the Basic Safety Standards (BSS) [EU 96] published by the European Council in 1996 and the Council Directive on health protection of individuals against dangers of ionising radiation in relation to medical exposure [EU 97] had to be transferred into national law within due time. In 2001 the new Ordinance for the Implementation of the Euratom Guidelines on Radiation Protection [STR 01] was published, which replaces the old Radiation Protection Ordinance [STR 89]. The Ordinance had been worked out by the Federal Government under participation of several Federal Ministries and of expert committees in charge. The draft was commented by associations and organisations taking influence on several regulations. Finally it was passed by the “Chamber” of the Federal States (Bundesrat) applying some changes.

The new German Ordinance adapts the European Directive to German law, covering the general principles but even giving more details in many fields of radiation protection. The scope of the BSS is laid down in Article 2:

1. This Directive shall apply to all practices which involve a risk from ionising radiation emanating from an artificial source or from a natural radiation source in cases where natural radio nuclides are or have been processed in view of their radioactive, fissile or fertile properties, namely:
 - a) the production, processing, handling, use, holding, storage, transport, import to and export from the Community and disposal of radioactive substances;
 - b) the operation of any electrical equipment emitting ionising radiation and containing components operating at a potential difference of more than 5 kV;
 - c) any other practice specified by the Member State.
2. In accordance with Title VII it shall also apply to work activities which are not covered by paragraph 1 but which involve the presence of natural radiation sources and lead to a significant increase in the exposure of workers or members of the public which cannot be disregarded from the radiation protection point of view.

3. In accordance with Title IX it shall also apply to any intervention in cases of radiological emergencies or in cases of lasting exposure resulting from the after-effects of a radiological emergency or a past or old practice or work activity.
4. This Directive shall not apply to exposure to radon in dwellings or to the natural level of radiation, i.e. to radio nuclides contained in the human body, to cosmic radiation prevailing at ground level or to aboveground exposure to radio nuclides present in the undisturbed earth's crust.

This scope certainly is much broader than the prescriptions important for the field of radiation protection in nuclear power plants. According to the scope of this workshop on occupational exposure in nuclear power plants – and as the BSS most probably will be quite familiar to all of you – after a short general overview on relevant contents of the German Ordinance, this presentation will focus on the main issues important in the operation of NPP and especially on some areas which may give rise to necessary changes caused by the new Ordinance.

A. Overview on the radiation protection ordinance

In its new form, the Ordinance covers the following main sections:

General regulations

This section covers the purpose and scope of the Ordinance and the definitions. For instance in this part the conceptual distinction between “practices” involving radiation from (in general) artificial sources and “work” activities with the presence of radiation from natural sources is defined.²

Protection of man and environment against radioactive substances or ionising radiation due to practices

This section covers essentially:

- Basic principles of radiation protection and general limits, as e.g. justification, limitation and optimisation.
- Authorisation for
 - possessing, handling of radioactive substances and for equipment generating ionising radiation;
 - transport of radioactive substances;
 - transboundary shipment of radioactive substances;

2. The Ordinance distinguishes “Tätigkeiten” and “Arbeiten”. “**Tätigkeiten**” are defined as operation of facilities to generate ionising radiation, as addition of radioactive substances during production or activation of such products, and as all other activities which may increase exposure or contamination during handling artificial radioactive substances or natural radioactive substances, if handled because of their radioactivity. This will be translated as “practices”. “**Arbeiten**” are defined as increasing the exposure due to natural radiation without being ‘practices’ as defined above. This will be translated as “work activities”.

- medical research;
 - undertakings providing occupationally exposed personnel to operators;
 - clearance.
- Type approval regarding apparatus with radioactive sources or generating ionising radiation.
 - Exemptions from the need of a license or approval.
 - For these fields the criteria and requirements for practices needing no authorisation as well as the conditions and prerequisites for an authorisation are laid down.
 - Requirements for practices
This section covers the necessary qualification and training in radiation protection, organisation of radiation protection, the protection of persons in radiation areas as well of man and environment against exposure, the protection against significant safety related events, dose limits, work-related medical provisions for occupationally exposed persons, procedures for radioactive waste and other requirements covering e.g. storage and safeguarding of radioactive substances, measuring devices, handing over of radioactive substances including documentation and accounting.

Protection of man and environment against natural sources of radiation due to work activities

For the first time (apart from the nuclear fuel cycle) in Germany, regulations regarding exposures to enhanced natural radioactivity are incorporated into the ordinance:

- The basic requirements regarding dose limitation and restriction.
- The requirements in case of terrestrial radiation at work-places:
Criteria for reporting if the exposure of personnel may exceed 6 mSv/year, and protective measures are dealt with. The dose limits are the same as set for occupationally exposed personnel performing practices (see Chapter 3).
- Protection of the public against naturally occurring radioactive material:
Criteria and requirements regarding radioactive residues - resulting from work activities with natural radioactive material – which need surveillance are defined. It also binds the producer to remove these residues after the end of his work activities to make this area usable for other purposes. The criterion for further unconditional use is a dose limit of 1 mSv/year to the individual.
- Cosmic radiation:
Regarding the protection of flight personnel against cosmic radiation the scope of surveillance and dose assessment (ambient dose calculation), the dose limits and the requirements for documentation are defined. Also in this field the dose limits are aligned to the limits set for occupationally exposed personnel in practices.

Protection of the consumer against the addition of radioactive substances to products

The products for which deliberate addition of radioactive substances or activation is forbidden and the requirements for those products for which deliberate addition or activation will need an authorisation are defined. The prerequisites for an authorisation are connected to activity limits

derived from exemption and clearance values. Transboundary shipment of such products also needs authorisation.

Common requirements

Requirements common for practices and for work activities are defined, in particular

- exposures to be taken into account;
- national dose registry;
- interim regulations regarding continued practices according the old ordinance and demarcation against other regulations.

Much information and data is added to the Ordinance in the form of enclosures covering e.g. the definition of practices not needing reporting or authorisation and the necessary data to be forwarded for application of an authorisation.

The most important ones are:

- the tables defining the exemption limits, clearance levels, values of the surface contamination;
- requirements for the clearance of radioactive material;
- parameters and maximum concentrations for radio nuclides discharged from facilities into air or into water including calculation procedures.

B. Regulations important for NPP-operation and issues to be solved

The issues with some relevance for nuclear power plants (NPP) will cover the new dose limits, other reference levels and the clearance to be met by the utilities during operation or dismantling, and the protection measures for outside workers. Emphasise is given to the need of changes in organisation, planning and personnel management in comparison to the regulations according to the “old” Ordinance [STR 89].

Dose limits

Regarding these limits, new dose limits to the public as well as to occupationally exposed personnel have to be considered. In both fields implications may be expected. The limits in this presentation mainly will be discussed taking the effective dose as a reference, as this is the reference chosen also in the BSS with additional limits for the equivalent dose for the eye’s lens and the skin. In Germany, due to historical reasons and with the argument to limit any potential higher exposure of some organs, additionally limits for the equivalent doses of organs have to be met for the public and for occupationally exposed persons. These will not be addressed in detail, but are summarised in a table in the annex to this paper (see Chapter 6.1).

Doses to the public

The dose limit for the effective dose to persons of the public has been set to 1 mSv/year according to the BSS. Additionally there are dose limits laid down for the (operational) discharges from facilities which call for keeping the effective dose of members of the public below 0,3 mSv/year, with additional limits set for organ doses (see Table 3-1). These limits have to be met outside the facility's area (Betriebsgelände). This area is not necessary limited by the site's fence, but defined as the developed site where the facility is located and to which its responsibility for radiation protection is empowered to limit access and residence time.

Table 3-1. **Dose limits to protect members of the public**

Members of the public	
General limit, effective dose	1 mSv/y
(Sum of exposure due to direct radiation and discharges; for direct radiation, whole stay for 1 year in general)	
Dose limits for radioactive discharges	
Effective dose	0,3 mSv/y
Organ dose: ovary, uterus, bone marrow	0,3 mSv/y
Organ dose: colon, lung, stomach, bladder, breast, liver, thyroid	0,9 mSv/y
Organ dose: bone surface, skin	1,8 mSv/y
(For discharges into the air or water, respectively)	

Compared to the old Radiation Protection Ordinance /STR 89/, the dose limits for the discharges have not been changed. The value of the general limit, which had been 1,5 mSv/year has been replaced by 1 mSv. For calculation, a steady stay all over the year (8760 hours) has to be taken into account, if no reliable data for the residence time of persons are available.

To allow for the superposition of the exposure due to discharges and to direct radiation, as a consequence the dose rate from direct radiation at the area of the site needs to be evaluated thoroughly. Theoretically the direct radiation from the site should be restricted to about 0,4 mSv/year with a mean dose rate of about 0,05 μ Sv/h, if the utility can not assure time limitations for the presence of persons and doses from releases would be close to the limits, which is not the case. The fact, that actually the plants only exhaust the discharge limits to less than one percent hardly will help to increase the dose rates from direct radiation in practice.

In practice higher dose rates are possible by considering realistic residence times; but in some cases additional measures may be necessary as for instance improved shielding in critical areas or special planning, if e.g. the site is used for interim storage of radioactive material.

For companies which will detach personnel to work as occupationally exposed persons (outside workers or contracted personnel) in nuclear facilities, the reduced 1 mSv/year limit also implies more clarification and may have a consequence: An authorisation has to be applied for to detach personnel, if the effective dose of the person may exceed 1 mSv/year. Dose limits and radiation protection measures for this personnel apply in the same way as for utility personnel (see tables 3-4 and 3-5). The outside company has to equip each worker with a registered radiation passport covering all exposures for the worker.

Delineation of areas with potential exposures on site

The borders of the site area to some extent may be considered the „borderline” between members of the public and the group of persons exposed by their practices on site. The new Ordinance applies for lower thresholds for areas with potential exposures or dose rates(see Table 3-2). As an overview, Figure 3-1 gives an schematic presentation of the areas defined on site and the integration into the company’s and the public area.

Figure 3-1. **Definition of areas on- and off site; the dashed line defines the company’s area, which allows access of the public, but where the company has the power to restrict the access of the public to less than 1 year. Residence times indicated are applied if no data are available, but smaller values may be granted, if they can be proven**

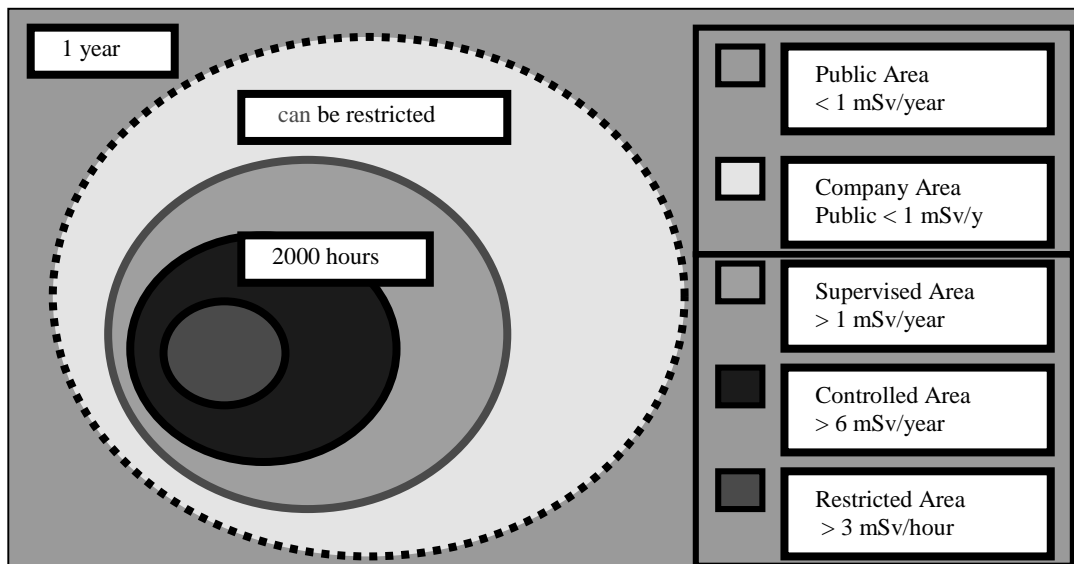


Table 3-2. **Delineation of radiological areas with potential exposures beyond 1 mSv/year effective dose (Organ doses according to BSS see table in the annex)**

“Supervised area”	
Effective dose	> 1 mSv/y
“Controlled area”	
Effective dose	> 6 mSv/y
“Restricted area”	
Dose rate	> 3 mSv/h

The “restricted area” is a specific German criterion requiring additional access and monitoring measures; this requirement is unchanged from the old Ordinance.

For supervised and controlled areas a working/residence time of 40 hours a week and 50 weeks per year is assumed resulting in a theoretical work time of 2 000 hours, if no other conditions can be proven.

Practical consequences may result from

- the limitation of effective dose to 1 mSv/year on the company's area (with reduced residence time of 2000 h) for persons working in this area. This issue will be addressed in the chapter on dose limits for occupationally exposed personnel;
- The reduction of the dose threshold for the supervised area from formerly 5 mSv/year to the new value of 1 mSv/year;
- the reduction of the dose threshold for the controlled area from formerly 15 mSv/year to the new value of 6 mSv/year.

Table 3-3. Dose rates derived from dose limits (if no exposures due to discharges are considered), assuming 2000 hours residence time

Public areas around the company's area (total dose rate)	< 0,11 µSv/h
Company's area, not radiologically surveyed	0,11 µSv/h to < 0,5 µSv/h
Supervised area	0,5 µSv/h to < 3 µSv/h
Controlled area	> 3 µSv/h (to 3 mSv/h)
Restricted area	> 3 mSv/h

Reducing the limit for the controlled area should not imply severe modifications in constructive areas of the plants, as normally in NPPs the borders of the controlled area are not limited by the dose rate, but by constructive boundaries. There are, however, areas where these reductions are to be considered, e.g. areas on site with temporary storage of radioactive waste or loaded fuel flasks, which may need some delimitation due to higher dose rate.

A working group of radiation protection specialists from the Vereinigung der Grosskraftwerkbetreiber (VGB, Association of large power plant operators) has developed some concepts [VGB 01] to transfer these regulations into practice.

For areas outside the company's influence:

If dose rates are below 0,11µSv/hour the dose limit is considered as being met; for dose rates near or above this value detailed considerations are recommended:

- In inhabited areas, designated living areas, mixed areas and camping areas steady residence (8 760 hours) has to be assumed.
- In commercial areas a maximum residence time of 2 000 hours (work time 40 hours 50 weeks) could be assumed.
- In rural and forestall areas a more limited residence time of 400 h may be assumed resulting in an average dose rate of 2,5 µSv/h.

As a superposition of doses for a single person in all areas considered can not be excluded, the sum over an assumed residence in all 3 areas should not be higher than 1 mSv. Additionally an exposure due to operational discharges needs to be considered in the resulting dose.

For areas inside the company's influence, this means in the company's area,

- the existing delimitation of supervised and controlled areas basically need not to be changed;
- care should be taken that outside of a controlled area doses higher than 1 mSv can not occur without being recognised and accounted;
- areas with continuous dose rates $> 0,5 \mu\text{Sv/h}$ outside the controlled area in any case will be restricted of or at least marked to assure that the dose limit of 1 mSv can be met for residence times of 2 000 h. In case of practices carried out in these areas, dosimeters have to be used or the dose has to be assessed by other ways. In some cases also restrictions of residence times may be applicable.

Evaluations in NPPs have shown that these procedures are practicable and compatible with the existing situation at the sites.

Further consequences to be considered will be discussed in the following chapter in the context of the dose limits to occupationally exposed persons on site.

Dose limits for personnel and for visitors in nuclear power plants

The personnel defined as "occupationally exposed personnel", other workers and visitors need to be considered.

Table 3-4. **Occupationally exposed personnel**

Categorisation of occupationally exposed personnel	
Category A	
Effective dose	$> 6 \text{ mSv/y}$
Organ dose: eye lens	$> 45 \text{ mSv/y}$
Organ dose: skin, hands, arms, feet, ankle	$> 150 \text{ mSv/y}$
Category B	
Effective dose	$> 1 \text{ mSv/y}$
Organ dose: eye lens	$> 15 \text{ mSv/y}$
Organ dose: skin, hands, arms, feet, ankle	$> 50 \text{ mSv/y}$

These dose thresholds follow the BSS. Compared to the old German regulation there are two main changes:

- the dose thresholds of the two groups of occupationally exposed personnel defining the category has been significantly reduced (5 mSv to 1 mSv for category B, 15 mSv to 6 mSv for category A Personnel);
- the former group of occupationally exposed personnel not categorised, but allowing exposure up to 5 mSv/year [STR 89] does no longer exist. In practice for this group a limit has been set to 1 mSv/year as to members of the public.

As a consequence in practice the groups have been “scaled down” with all persons potentially receiving doses beyond 1 mSv/year being designated at least to group B. As given in the BSS, the main difference between category A and B-personnel is the way of dose assessing and medical surveillance. Considering this difference, the “upgrading” of personnel from category B to category A in case of need might be possible, but will not be a practical solution due to the formalism and the time necessary.

Though seeming of less importance, the lowering of the dose threshold for personnel of category B (or the “dose limit” for non categorised occupationally exposed personnel) will afford some consideration on site for the utility personnel and also for a certain group of contracted personnel.

- According to the old Radiation Protection Ordinance for utility personnel and other persons working on site not dedicated to routinely work linked exposure, as e.g. clerical personnel, guards, ... , could be exposed to a maximum of 5 mSv/year as uncategorized occupationally exposed personnel. This procedure is no longer possible with the need either to assure that this group of personnel will meet a dose limit of 1 mSv/year or to define them as occupationally exposed personnel of category B. As mentioned above, the utilities tend to take care that the dose limit of 1 mSv/year can be met for this group, e.g. by assuring dose rates to less than 0,5 µSv/h in areas on site which are accessible for this personnel.
- For a certain group of contracted personnel detached from entities the situation is quite close, as it was possible to apply up to 5 mSv/year to persons not being occupationally categorised. Under the new Ordinance the plant needs to handle this group of personnel not to receive a dose exceeding 1 mSv/year. This will necessitate some special formal arrangement with the contractor and some special dose considerations or work restrictions for those workers. Or, as an alternative these persons would need to be assigned at least to category B with the additional consequence that the contractor’s-company would need a special license and has to file the exposures in a radiation passport.

Visitors as members of the public will have to keep a dose limit of 1 mSv/year. To be sure to meet this regulation, utilities have considered a lower internal limit to be met for visitors in the plant.

The dose limits for occupationally exposed personnel are shown in Table 3-5 according to the Radiation Protection Ordinance which applies additional dose limits for organ doses compared to the BSS.

Table 3-5. Dose limits for occupationally exposed personnel

Effective dose	20 mSv/y*
Organ dose: eye lens	150 mSv/y
Organ dose: skin, hands, arms, feet, ankle	500 mSv/y (each)
Organ dose: ovary, uterus, bone marrow	50 mSv/y (each)
Organ dose: thyroid, bone surface	300 mSv/y (each)
Organ dose: colon, lung, stomach, bladder, breast, liver, oesophagus	150 mSv/y (each)

* The authorities may permit a limit of 50 mSv/a with the limitation of 100 mSv over the period of the following 5 years.

The 20 mSv/year limit has been adopted as common practice, but the exception of 50 mSv/year with an average of 20 mSv/year over 5 years may be granted by special agreement of the authorities. Actually the data show, that for utility personnel in nearly all German NPPs the new limit should not be a problem. For contracted personnel in some areas this may afford special personnel deployment planning to keep the 20 mSv/year-limit, although plant specific data also for this personnel show that the contribution to the exposure of contracted personnel in most of the NPPs is quite low.

Additional dose limits are defined for special groups of persons due to age, training needs or due to sex to protect unborn children and breast fed babies.

Table 3-6. Dose limits defined for the occupational exposure of special groups of the public (effective dose presented only)

Persons less than 18 years old	
Effective Dose	1 mSv/y
Trainees, students, aged 16 to 18, by agreement of the authorities	
Effective Dose	6 mSv/y
Women who may become pregnant	
Uterus Dose	2 mSv/month
Women who are pregnant	
Internal and external dose over time of pregnancy no incorporation -> no employment in controlled areas of NPPs with presence of radioactivity	1 mSv
Career dose for occupationally exposed persons	
Effective dose over sum of all work time	400 mSv

For persons less than 18 years, for trainees aged 16 to 18 years and for the protection of the unborn child these limits follow the guidance of the BSS.

Special German regulations are the limits set for women who may become pregnant to assure some additional protection in case of pregnancy until it is recognised and the career dose limit which was taken from the old Radiation Protection Ordinance, which, however, – as a well known regulation – will not cause any conflicts and which has some exception regulation if personnel should exceed this limit.

The limit for women who may become pregnant will need special dose assessment. Additionally special information is necessary for this group. There are prescriptions to inform about the hazards of activity by internal contamination. Additionally it is prescribed that for women who have informed the utility that they are pregnant or breast feeding the work conditions have to be set in a way that incorporation of radionuclides is excluded.

As a practical proposal, additionally to the necessary information of special groups of the personnel, in [VGB 01] it is proposed for all workers to set an internal limit of about 80% of the official dose limit and additionally to set the daily dose budget to about 10% of the dose limit to be able to plan and manage dose development of the staff in due time.

Dose limits for specially authorised exposures of workers category A

To complete the field of dose limits, two additional cases according to Article 12 and Article 52 of the BSS, are to be listed.

Table 3-7. **Special dose limits set for special work**

Special work (dose limit)	
Effective dose	100 mSv/y
Organ dose: eye lens	300 mSv/y
Organ dose: skin, hands, arms, feet, ankle	1000 mSv/y
Agreement of authorities and personnel necessary	

Table 3-8. **Special dose levels recommended for life saving actions**

Life saving actions (reference level, no limit)	
Effective dose (once a year)	100 mSv
Effective dose (once in life time)	250 mSv

Details of regulations for specially authorized exposures are as prescribed by Article 12 BSS. It has to be stressed that in the case of life saving actions the dose values indicated are not considered as limits, but more as a recommendation.

To some extent, both regulations cover cases which are more hypothetical. For the operation of a NPP this should be an exception and should not create special restrictions. For the utilities it is, however, considered to derive a concept for information of volunteers.

Exemption values

In Table A to its Annex I the Basic Safety Standards lay down exemption values defined as activities and as specific activities for a large number of nuclides, which give the criteria for exemption from reporting or authorisation. This table has been integrated into the German Ordinance [STR 01] and has been expanded to a large number of additional nuclides derived by the National Radiation Protection Board (NRPB) [NRPB 98].

An example is given in Table 3-9 showing the full scope of additional data integrated which cover regulations dealt with in the next chapters. The data for ⁶⁰Co are presented as an example for the criteria chosen.

Table 3-9. Regulation contained in the new German Ordinance regarding exemption, surface contamination and clearance. Additional conditions and regulation have to be considered for these regulations (see Chapters 6.2 and 6.3 in the annex of this paper)

Nuclide	Exemption Bq	Exemption Bq/g	Surface cont. Bq/cm ²	Solid/Liquid Bq/g	Building rubble Bq/g	Soil Bq/g	Building re-use Bq/cm ²	Solid/Liquid Bq/g	Building dism. Bq/cm ²	Metal scrap Bq/g	Half-life
⁶⁰ Co	1E5	1E1	1	0,1	9E-2	3E-2	4E-1	4	3	0,6	5,3 y
	Exemption values		Clearance values unconditional release				Clearance values conditional release				

Basically the large number of nuclide specific values defined afford the superposition of nuclides which is done by a the summation formula $\sum_i \frac{\text{value considered or measured}_i}{\text{value from table}_i} \leq 1$

to account for the relative normalised sum over all relevant nuclides i for the item to be evaluated as e.g. activity, specific activity or surface contamination. In practice this will call for some effort, as in a nuclear power plant a specific nuclide vector prevails. This nuclide vector needs to be evaluated and implemented into the measurement. In practice this will be done by specific calibration of the measurement devices. With time, the nuclide vector has to be checked for changes due to operation or changes of conditions and in case re-calibration has to be performed.

Surface contamination

The lists values for surface contamination which will be used as a basis for the clearance of material (see Chapter 3.4) and to set maximum values for the surface contamination in different areas including tools and apparatus on and off site. As mentioned in Chapter 3.2, the summation formula has to be applied for nuclide vectors.

Table 3-10. Multipliers to be applied to calculate the limits for surface contamination in different areas on and off site

Area or material	Multipliers to be applied to the value given for surface contamination
Controlled area	100
Supervised area	10
Company area	1
Material to be released from controlled area (specific activity to be checked too)	1

Clearance for re-use, recycling and disposal

Criteria for the release of usually slightly contaminated or activated material from the requirements for radioactive substances have been defined in the Radiation Protection Ordinance. Related clearance values for different ways of recycling or re-use and disposal have been integrated into the Ordinance as can be taken from Table 3-9. As possible pathways the unconditional use of the material, and the conditional use, understood as disposal in a refuse disposal site, incineration or conditional recycling by melting are considered. This consideration and data are also related to the contamination of work-places, clothes and material on site and the transfer of such material (e.g. tools, instruments, ...) from the controlled area for use outside controlled areas.

Regarding the clearance of radioactive material for unconditional or conditional use, an approval of the authorities will be necessary. It will be granted, if as a consequence of the release a member of the public only may receive a dose of about 10 $\mu\text{Sv}/\text{year}$. The authority can assume that this dose criteria will be met, if the values defined in Table 1 of Annex III of the Ordinance (see example in Table 3-9) are not exceeded and some additional conditions are fulfilled, which are defined in several sections of Annex IV of the Ordinance. Some information on the values laid down and the additional conditions and considerations can be found in the annex of this paper (see Chapter 6.3). All radioactive material not meeting these conditions for unconditional or conditional release certainly will have to be conditioned and disposed off as radioactive waste.

In the new Ordinance the first time in Germany a complete set of data has been laid down to regulate the release of radioactive material, building and areas from the surveillance due to the Atomic Law or the Radiation Protection Ordinance. Certainly this regulation has some advantage, as it clarifies the procedures, levels and conditions and so to some extent hopefully simplifies the formalisms and the handling of this slightly radioactive material.

For solid material with measurable surfaces, both levels for surface contamination and specific contamination have to be met, putting some effort into the measurement with large surfaces. Additionally material properties may cause some problems: for material with low specific weight, as e.g. insulation material, the masses may be significantly lower than e.g. for steel, which may give a higher specific activity and a higher volumetric relation.

In practice special measuring chambers of different sizes are able to proof the specific activity levels of the material to be released. This holds for operating plants and for plants under decommissioning. Practice also has been gained in the release of buildings and of soil areas by contamination monitoring and in situ-contamination monitoring of large areas and also of contaminated soil by specific activity measurement e.g. in Greifswald.

Though some effort needs to be taken in some areas, the system has proven its applicability and effectiveness in practice to handle the release of material with limited content of radio nuclides.

Summary

The presentation gave a short overview of the new German Ordinance for the Implementation of the Euratom Guidelines on Radiation Protection. As it was impossible to go into all details of this new Ordinance, after the short survey on its structure and main contents the presentation concentrated on two issues which may be important as well for the operation and for the dismantling of nuclear facilities, focussing on dose limits and on clearance levels. Whereas the dose limits have undergone some important changes and reductions of values, guidance on the release (clearance) of

low radioactive material either for unconditional use or for conditional use by recycling or by disposal has been implemented as a new, but comprehensive part into the new Ordinance.

Though some effort has to be taken by the nuclear power plants, it can be recognised that the implementation of the new regulation should not cause severe restrictions for the utilities, but in some cases may increase cost and in some cases also the amount of formalism necessary.

References

- [1] [EU 96] Council Directive 96/29/Euratom of 13 May 1996 laying down basic safety standards for the protection of the health of workers and of the general public against the dangers arising from ionizing radiation. Official journal No. L 159, 29.06.1996 p. 1 ff.
- [2] [EU 97] Council Directive 97/43/EURATOM of 30 June 1997 on health protection of individuals against the dangers of ionizing radiation in relation to medical exposure, and repealing Directive 84/466/EURATOM. Official journal No. L 180, 09.07.1997 p. 22 ff.
- [3] [NRPB 98] Mobb, S.F., M.P. Harvey Exempt concentrations and quantities for radio nuclides not included in the European Basic Safety Standards Directive. NRPB-R 306, Approved December 1998, Published April 1999. ISBN 0 85951 429 3.
- [4] [STR 89] Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung-StrlSchV), in der Fassung der Bekanntmachung vom 30. Juni 1989 BGBl. I 1989 S. 1321, ber. S. 1926.
- [5] [STR 01] Verordnung für die Umsetzung der EURATOM-Richtlinien zum Strahlenschutz vom 20. Juli 2001 BGBl. I 2001 S. 1714.
- [6] [VGB 01] Praktische Umsetzung der Strahlenschutzverordnung vom 1. August 2001 in Kernkraftwerken Interpretationshilfen und Empfehlungen des Arbeitskreises "Praktischer Strahlenschutz" der Vereinigung der Gross-Kraftwerksbetreiber (VGB). Stand August 2001.

Annex

Dose limits defined in the new ordinance

Table 6-1. **Protection of the public and of occupationally exposed persons**

Reference	Dose Limit
Protection of the public	
Members of the public, dose limits	
Effective dose (Sum of exposure due to direct radiation and discharges of radioactivity into the air and into water)	1 mSv/y
Organ dose: eye lens	15 mSv/y
Organ dose: skin	50 mSv/y
Dose limits for radioactive discharges to protect the public	
Effective dose	0,3 mSv/y
Organ dose: ovary, uterus, bone marrow	0,3 mSv/y
Organ dose: colon, lung, stomach, bladder, breast, liver, thyroid	0,9 mSv/y
Organ dose: bone surface, skin	1,8 mSv/y
Limits for discharges into the air and into water, respectively	
Protection of occupationally exposed persons	
Categorisation of occupationally exposed personnel	
Category A	
Effective dose	> 6 mSv/y
Organ dose: eye lens	> 45 mSv/y
Organ dose: skin, hands, arms, feet, ankle	> 150 mSv/y
Category B	
Effective dose	> 1 mSv/y
Organ Dose: eye lens	> 15 mSv/y
Organ Dose: skin, hands, arms, feet, ankle	> 50 mSv/y
Dose limits for occupationally exposed personnel	
Personnel Category A	
Effective dose	20 mSv/y*
Organ dose: eye lens	150 mSv/y
Organ dose: skin, hands, arms, feet, ankle	500 mSv/y (each)
Organ dose: ovary, uterus, bone marrow	50 mSv/y (each)
Organ dose: thyroid, bone surface	300 mSv/y (each)
Organ dose: colon, lung, stomach, bladder, breast, liver, oesophagus	150 mSv/y (each)

* The authorities may permit a limit of 50 mSv/a with the limitation of 100 mSv over the period of the following 5 years.

Table 6-2. Protection of special groups of occupationally exposed persons

Reference	Dose Limit
Persons less than 18 years old	
Effective dose	1 mSv/y
Organ Dose: eye lens	15 mSv/y
Organ Dose: skin, hands, arms, feet, ankle	50 mSv/y
Trainees, Students	
Effective dose	6 mSv/y
Organ dose: eye lens	45 mSv/y
Organ dose: skin, hands, arms, feet, ankle	150 mSv/y
Women who may become pregnant	
Uterus dose	2 mSv/month
Women who are pregnant	
Internal and external dose over time of pregnancy	1 mSv
Career dose for occupationally exposed persons	
Effective dose over sum of all work time	400 mSv
Specially authorized exposures, (dose limits)	
Effective dose	100 mSv/y
Organ dose: eye lens	300 mSv/y
Organ dose: skin, hands, arms, feet	1000 mSv/y
Agreement of authorities and personnel necessary	
Live saving actions, (reference dose, no limit)	
Effective dose (once a year)	100 mSv
Effective dose (once in life time)	250 mSv

Table 6-3. Delineation of areas with potential exposures on site

Reference	Dose Limit
“Supervised area”	
Effective dose	> 1 mSv/y
Organ dose: eye lens	> 15 mSv/y
Organ dose: skin, hands, arms, feet, ankle	> 50 mSv/y
“Controlled area”	
Effective dose	> 6 mSv/y
Organ dose: eye lens	> 45 mSv/y
Organ dose: skin, hands, arms, feet, ankle	> 150 mSv/y
“Restricted area”	
Dose rate	> 3 mSv/h

Table 6-4. **Incident reference levels for planning and design of technical protective measures against incidents (To be applied for the design of NPP, storage of nuclear fuel, and facilities of the Federal Government for confiscation and final disposal of radioactive waste)**

Reference	Dose level
Effective dose	50 mSv
Organ dose: thyroid, eye lens	150 mSv
Organ dose: skin, hands, arms, feet, ankle	500 mSv
Organ dose: ovary, uterus, bone marrow (red)	50 mSv
Organ dose: bone surface	300 mSv
Organ dose: colon, lung, stomach, bladder, breast, liver, oesophagus	150 mSv
Doses are to be calculated according to Guideline and cover exposure over 50 years including ingestion	

Overview of release criteria of radioactive material for unconditional use, disposal and recycling

This annex gives a short overview over the contents of the Table 1 in Annex III of the Ordinance. The table heading and one entry for Co 60 are displayed for orientation.

Nuclide	Exemption Bq	Exemption Bq/g	Surface cont. Bq/cm ²	Solid/Liquid Bq/g	Building rubble Bq/g	Soil Bq/g	Building re-use Bq/cm ²	Solid/Liquid Bq/g	Building dism. Bq/cm ²	Metal scrap Bq/g	Half-life
⁶⁰ Co	1E5	1E1	1	0,1	9E-2	3E-2	4E-1	4	3	0,6	5,3 y
	Exemption values			Clearance values unconditional release				Clearance values conditional release			

The columns of the table contain:

- **Radio nuclides**, to some extent also special chemical forms of a specific nuclide.
- **Exemption values** for the nuclides defined as activities and specific activities.
- Surface contamination levels to define:
 - (by certain multipliers) the maximum surface contamination for work-places and clothing in the controlled area, in the supervised area and outside;
 - the maximum surface contamination for objects which are transferred from the controlled area for use outside such areas (also specific activity criteria have to be met, see below);
 - the maximum surface contamination of solid waste for disposal and metal scrap for recycling (also specific activity criteria have to be met, see below) if the material has some fixed defined surface.

- **Specific activities for liquids and solid materials** (except building rubble and excavated material with masses > 1 000 t per year) which give the specific activity limit:
 - for objects which are transferred from the controlled area for use outside such areas (also surface contamination have to be met, see above);
 - for an unconditional release of solid material (criteria for surface contamination have to be met, see above);
 - for unconditional release of liquids.
- **Specific activities for building rubble and excavated material** of more than 1 000 t/year for unconditional release of this material.
- Specific activities for soil areas for unconditional use.
- **Surface contamination levels for building surfaces** for the release of buildings for unconditional use.
- **Specific activities for liquids and solid materials** (except building rubble and excavated material with masses > 1 000 t per year) which give the specific activity limit for conditional release:
 - for disposal of the material (surface contamination limits have to be met) e.g. in a refuse disposal site;
 - incineration of liquids.
- Surface contamination levels for building surfaces for the dismantling of buildings.
- **Specific activities for metal scrap for recycling** (surface contamination levels have to be met, see above).

Overview of additional conditions to be met and considered in the release of radioactive material in unconditional release, for disposal or recycling as defined in Annex IV of the Ordinance

The important statements of Annex IV of the Ordinance are listed to explain philosophy and boundary conditions for the release of radioactive material. It is not an exact translation of the respective annex of the Ordinance, if necessary the original document should be referred to:

- General conditions
 - If the surface of the material allows a surface contamination measurement, it has to be shown by measurement, that the surface contamination limits and the activity concentration limits are met (authority may allow other ways of proof).
 - The averaging mass for the measurement of the activity concentration shall be not significantly larger than 300 kg.
 - The averaging area for contamination measurement may be up to 1 000 cm².
 - If several radionuclides are present, a summation formula for the normalised contributions shall be applied for the specific activity and the surface

contamination; nuclides need not to be considered, if the contribution of the neglected nuclides is less than 10% of the normalised sum.

- For radionuclides in radioactive equilibrium, the daughter radio nuclides listed in a specific table may be neglected in the summation.
 - For those nuclides which are not listed in the Table III, the values have to be calculated for the specific case; for nuclides with half lives less than 7 days or in case of small masses, the specific activity of the exemption level may be used for the release of solid or liquid material.
- Unconditional release
 - The unconditional release covers no further commitments regarding the future use.
 - For building rubble and excavated material the values for solid material may be used, for unconditional release, if the mass is smaller than 1 000 t/year.
 - Release for disposal
 - It is a prerequisite that the material will be implemented into the refuse disposal site without biological or chemical pre-processing or into an incineration plant and that a use of the material outside the refuse disposal site or incineration plant will be excluded.
 - The values given for “building dism.” do not apply for building rubble and excavated material if the mass may be larger than 1 000 t/year.
 - Release of buildings
 - Measurements should be applied at the non destructed building and may be performed applying a spot check procedure.
 - the averaging area may be up to 1 m².
 - After release of a building arising building rubble e.g. from dismantling does not afford a separate release procedure.
 - In case of activation of the material resulting in a volumetric activity distribution, further prescribed criteria have to be applied.
 - Release of contaminated soil areas
 - The averaging area of the contaminated surface may be up to 100 m².
 - Only such contamination has to be considered, which was caused by the facility under consideration.
 - If no reference levels are defined, the proof of dose limitation to the public has to be performed by dose calculation based on contamination measurements.
 - The specific activity clearance level for soil may be re-calculated into a surface contamination level (surface level = specific level * soil density * mean penetration depth).

- Release of building rubble and excavated material
 - The values given for building rubble and excavated material hold for material from practices in facilities in operation or from dismantling of a facility or its parts, if the prerequisites for clearance measurement at the intact building are not met.
 - The averaging mass for building rubble may be up to 1 ton (1 Mg); the authority may allow higher masses.
- Release of metal scrap for recycling
 - It is a prerequisite for this release path that the scrap released will be molten.

The values defined are not valid for compounds of metallic and non-metallic material.

IMPLEMENTATION OF THE BASIC SAFETY STANDARDS DIRECTIVE IN THE UK

W. Bines

OBE, FSRP

Health and Safety Executive, HD B Ionising Radiation Policy Unit, England

Abstract

Implementation of the European Council BSS Directive 96/29/Euratom in the UK is not achieved through any one piece of legislation (though the majority of the provisions are implemented by the Ionising Radiations Regulations 1999) but by a mosaic of provisions, supported by codes of practice, non-statutory guidance and administrative arrangements. The paper describes some of the features of UK occupational radiation protection and the reason for the apparent differences between the UK and other EU Member States in their approach to agreeing the precise provisions of European legislation.

Introduction

In the United Kingdom no single legal instrument gives effect to European Council Directive 96/29/Euratom (the “BSS Directive”, ref. 1) but several Acts and Regulations together, plus administrative arrangements, achieve implementation. The principal pieces of legislation are: the Ionising Radiations Regulations 1999 (IRR99) and Approved Code of Practice; the Radiation (Emergency Preparedness and Public Information) Regulations 2001; the Nuclear Installations Act 1965; the Medicines Act 1968 and the Medicines (Administration of Radioactive Substances) Regulations 1968 and associated Regulations and Orders; the Radioactive Substances Act 1993 and associated Regulations and Orders; the Food Safety Act 1990; the Environment Act 1995; the Food and Environment Protection Act 1985; and the Air Navigation Order 2000. Devolution has further complicated the picture in that Scotland and Wales, as well as Northern Ireland, in some cases must now make their own implementing legislation.

The UK approach to statutory interpretation

The UK is well known, during negotiations on draft directives, for seeking to insert qualifying phrases such as “as far as is reasonably practicable” and “where appropriate” into texts where other Member States would be content with, or even wish to see, absolute requirements. Until the reason for this approach is understood, it may be seen as attempting to reduce the force or the level of standards and provisions. However, the reason why UK negotiators have to take this line is because our courts interpret the law as written and have no discretion to decide appropriateness unless it is inbuilt. This means that British courts adopt a strict literal approach to interpretation, unlike their counterparts in some other continental jurisdictions where courts (or competent authorities) can decide appropriateness even when the requirement is apparently an absolute without any exception. British

judges would certainly not read in words such as “as appropriate” where those were lacking. Our law has also developed on a case-by-case basis and strict adherence to precedent is a feature.

Specific features of occupational radiation protection and enforcement

Some specific features of UK occupational radiation protection legislation and enforcement are:

- approved codes of practice and guidance;
- reasonably practicable;
- investigation levels;
- dose limits;
- criteria for recognition of approved dosimetry services and qualified experts;
- enforcement tools.

These are considered in more detail in the following paragraphs.

Approved Codes of Practice and guidance

In the UK, a Code of Practice approved by the Health and Safety Commission, with the consent of the Secretary of State, has a special legal status. It gives practical advice on how to comply with the law and an employer who follows the advice will be doing enough to comply with the law (in respect of those specific matters on which the Code gives advice). Employers may use alternative methods to comply with the law but, in that case, if they are prosecuted they will need to show that those other methods achieved the necessary compliance.

Non-statutory guidance often accompanies Regulations and Approved Codes of Practice [(these days often in the same document, for convenience, as in the Approved Code of Practice supporting the Ionising Radiations Regulations 1999 (Work with ionising radiation – Approved Code of Practice and guidance, L121, ref. 2)]. Following such guidance is not compulsory and employers are free to take other action. But employers who do follow the guidance on a specific legal requirement will normally be doing enough to comply with the law. Health and Safety Inspectors, who seek to secure compliance with the law, may refer to the guidance as illustrating good practice.

Much of HSE’s non-statutory guidance is freely available through the Ionising Radiation page of the Health and Safety Executive’s website (ref. 3).

Reasonably practicable

Some legal provisions may impose a duty on an employer without qualification, that is one that must always be carried out without exception. Others may require an employer to carry out a precautionary action “so far as is reasonably practicable”, or “where reasonably practicable”. “Reasonably practicable” is a narrower term than “physically possible” and implies a computation between the degree of risk and the sacrifice (in terms of money, time or trouble) involved in the measures necessary to avert the risk. If it can be shown that there is a gross disproportion between them, the risk being insignificant in relation to the sacrifice, the person upon who the duty is laid

discharges the burden of proving that compliance was not reasonably practicable. The computation must have been done before the incident complained of.

The competent authority may publish guidance on what it considers to be “reasonably practicable”. For example, regulation 8(1) of the Ionising Radiations Regulations 1999 requires a radiation employer to “take all necessary steps to restrict so far as is reasonably practicable the extent to which his employees and other persons are exposed to ionising radiation.” and regulation 8(2) establishes a hierarchy of control measures for this purpose (firstly engineered means, then supporting systems of work and lastly personal protective equipment), also qualified by reasonable practicability. In relation to exposure controls, paragraph 88 of L121 (ref. 2) says “Normally, it should be reasonably practicable to design control units for x-ray generators (and, where appropriate, radioactive source containers) to prevent unintended and accidental exposure.”. Similarly, in relation to warning devices, paragraph 101 says “Automatic warning devices should be reasonably practicable for most x-ray generators and some sealed sources.”

Investigation levels

The over-riding requirement of the Ionising Radiations Regulations 1999 is that employers must take all necessary steps to restrict, so far as is reasonably practicable, the extent to which their employees and other persons are exposed to ionising radiation. As part of this requirement employers must carry out an investigation when, for the first time in a year, an employee’s effective dose reaches **either** the level specified in the Regulations (15 mSv) **or** a lower level specified by the employer as being more appropriate for their practice. The employer could select different investigation levels for different sites or different groups of employees, where appropriate.

The purpose of this provision is to trigger a review of working conditions, to ensure that exposure is being restricted as far as reasonably practicable. The duty to carry out the investigation is placed on the actual employer of the person whose recorded dose has exceeded the investigation level. In most cases this will be an employer who is working with ionising radiation (a “radiation employer”). However, the employer might be a contractor (e.g. a scaffolding contractor or a cleaning company) working on various sites occupied by radiation employers. The investigation might have to take account of work with ionising radiation undertaken at all these different sites throughout the calendar year. Employers may wish to have arrangements for reviewing any unusually high doses, reported in dose summaries for classified persons (category A workers) by an approved dosimetry service or for other people entering controlled areas under written arrangements. Such arrangements would provide an early warning that an employee’s cumulative dose was approaching the investigation level and would allow the employer to take further measures to restrict exposure before a formal investigation became necessary.

Dose limits

The limit on effective dose, in the Ionising Radiations Regulations 1999 (ref. 4), for any employee aged 18 years of age or above is 20 mSv in any calendar year. However, the Regulations recognise that there may be some cases where, because of the special nature of the work undertaken by an employee, it may not be practicable to comply with this annual dose limit. This situation may arise where there are skilled tasks that need to be undertaken by key specialist staff, including foreign nationals. Where the employer can demonstrate that this is the case, the employer may apply the special dose limit of 100 mSv in five years (and no more than 50 mSv in any single year) to a named

employee. The choice of the five-year dose limit for any particular employee is subject to a number of preconditions (set out in Part 2 of Schedule 4 of the Regulations). These include:

- consultation with the radiation protection adviser (qualified expert) and with the affected employee(s) (and any appointed safety representatives);
- provision of information to the affected employee(s) and the approved dosimetry service; and
- giving prior notice to the Health and Safety Executive (HSE), which may (subject to appeal) over-ride the employer's decision and require the employer to revert to annual dose limitation for that employee.

Further conditions are imposed once the five-year dose limit has been applied to an employee, including:

- investigation of any suspected exposures exceeding 20 mSv in a calendar year and notify HSE (to check that the five-year dose limit will still be met);
- need to review whether five-year dose limit is still appropriate at least once every five years;
- restrictions on reversion to an annual dose limit for that employee; and
- recording and retention of the reasons for the five-year dose limit.

Approval of dosimetry services

Employers who designate employees as classified persons (category A workers) need to engage approved dosimetry services to undertake any necessary dose assessments, to open and maintain dose records and to provide relevant information from those records. Dosimetry services are approved by HSE (or a body specified by HSE) for one or more of the following specific purposes:

- measurement and assessment of whole-body or part-body doses arising from external radiation (notably x-rays, gamma rays, beta particles or neutrons);
- assessment of doses from intakes of specified classes of radionuclides;
- assessment of doses following an accident or other incident;
- co-ordination of individual dose assessments by other approved services, making, maintaining and keeping dose records, and provision of summary information; and
- assessment and recording of emergency exposures under the Radiation (Emergency Preparedness and Public Information) Regulations 2001.

An organisation may hold certificates of approval for more than one of these functions. The aim of approval is to ensure, as far as possible, that doses are assessed on the basis of accepted national standards and that dose records bring together all such dose assessments, helping employers check that doses are being kept as low as reasonably practicable and dose limits are not exceeded. For this purpose, HSE has published criteria for approval that must be met by a dosimetry service seeking approval or wishing to remain approved. The criteria Requirements for the approval of dosimetry services under the Ionising Radiations Regulations 1999 are available from HSE (ref. 5).

Recognition of qualified experts

Radiation protection advisers (RPAs) in the Ionising Radiations Regulations 1999 are the main “qualified experts” in UK legislation. Recognition of their capacity to act is in two parts: the competent authority sets criteria of core competence that all RPAs must meet, then there is a duty on the employer to consult and if appropriate appoint a **suitable** RPA. The criteria for recognition of RPAs is set out in the HSE Statement on radiation protection advisers (available on HSE’s website, ref. 3). RPAs must demonstrate competence against a list that is based on the basic syllabus for qualified experts contained in Annex 1 of the EC Communication on Directive 96/29/Euratom (ref. 6).

Enforcement tools

The ultimate purpose of the enforcing authorities is to ensure that duty holders manage and control risks effectively, thus preventing harm. The term ‘enforcement’ has a wide meaning and applies to all dealings between enforcing authorities and those on whom the law places duties (employers, the self-employed, employees and others).

Enforcement is distinct from civil claims for compensation and is not undertaken in all circumstances where civil claims may be pursued, nor to assist such claims. The enforcing authorities have a range of tools at their disposal in seeking to secure compliance with the law and to ensure a proportionate response to criminal offences. Inspectors may:

- Offer duty holders information, and advice, both face to face and in writing. This may include warning a duty holder that in the opinion of the inspector, they are failing to comply with the law.
- Where appropriate, serve:
 - improvement Notices (these may require an employer to undertake certain improvements by a specified date, while permitting work to continue, and may be used to tackle significant safety problems that nevertheless are not life-threatening);
 - prohibition Notices (a prohibition notice stops work in order to prevent serious personal injury. Information on improvement and prohibition notices should be made publicly available);
 - issue formal cautions (a formal caution is a statement by an inspector, that is accepted in writing by the duty holder, that the duty holder has committed an offence for which there is a realistic prospect of conviction); and
 - prosecute.

Giving information and advice, issuing improvement or prohibition notices and withdrawing or varying approvals, authorisations, licences or exemptions are the main means which inspectors use to achieve the broad aim of dealing with serious risks, securing compliance with health and safety law and preventing harm.

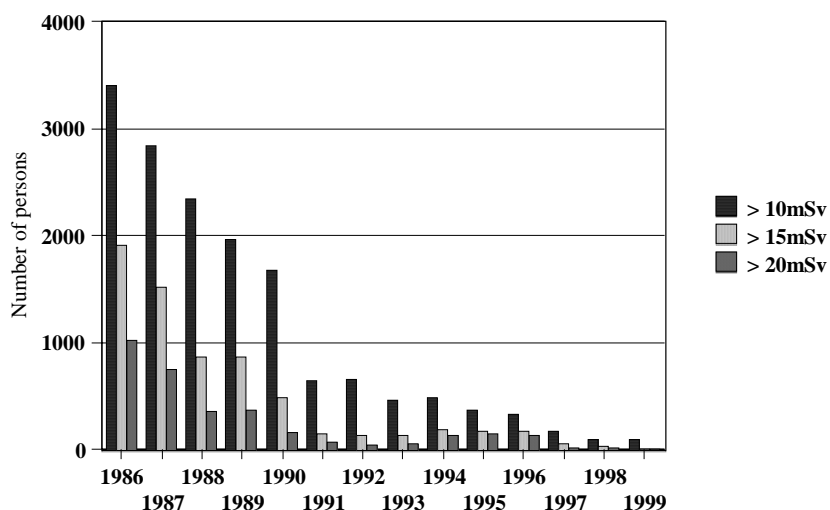
The UK’s Health and Safety Commission (HSC), of which the Health and Safety Executive is its executive arm, has issued a Policy Statement on Enforcement. In summary, HSC believes in firm but fair enforcement of health and safety law. This should be informed by the principles of proportionality in applying the law and securing compliance; consistency of approach; targeting of

enforcement action; transparency about how the regulator operates and what those regulated may expect; and accountability for the regulator's actions. The full Statement is available as a leaflet (HSC 15, ref. 7) or on the HSE website (ref. 3).

Conclusion

Implementation of the BSS Directive in the UK is well founded, being based on legal provisions supported by statutory and non-statutory guidance, administrative arrangements, case law and compliance enforcement. Taken with the effect of the legislation implementing the 1980 BSS Directive (ref. 8), the approach has been shown to be effective in reducing exposure levels (Table 1).

Table 1. **Number of classified (category A) persons exceeding certain levels of dose**
(source: HSE's Central Index of Dose Information)



References

- [1] 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 ionising radiation. *Official Journal of the European Communities* L159, 29.6.96, p. 1.
- [2] Work with ionising radiation. Ionising Radiations Regulations 1999. Approved Code of Practice and guidance L121 HSE Books* 2000 ISBN 0 7176 1746 7.
- [3] HSE ionising radiation website: www.hse.gov.uk/hthdir/noframes/iradiat.htm
- [4] The Ionising Radiations Regulations 1999, SI 1999/3232. The Stationery Office.

* HSE priced and free publications are available from HSE Books, PO Box 1999, Sudbury, Suffolk CO10 2WA, England. Tel.: (+44) (0) 1787 881165. Fax: (+44) (0) 1787 313995 Website: www.hsebooks.co.uk.

- [5] Requirements for the approval of dosimetry services under the Ionising Radiations Regulations 1999: Part 1 – External Radiations; Part 2 – Internal Radiations; Part 3 – Co-ordination and record keeping; Supplement on approval for emergency exposures during intervention. These are available from The Dosimetry Services Administrator, Health & Safety Executive, Physical Agents Unit, Magdalen House, Trinity Road, Bootle, Merseyside L20 3QZ Tel: 0151 951 4027; fax: 0151 951 4845; E-mail: brian.kemble@hse.gov.uk
- [6] Communication from the Commission concerning the implementation of Council Directive 96/29 Euratom laying down basic safety standards for the protection of the health of the workers and the general public against the dangers arising from ionising radiation. *Official Journal of the European Communities* C133, 30.4.98, p. 3.
- [7] Enforcement policy statement leaflet HSC15, HSE 2002. HSE Books.
- [8] Council Directive 80/836/ Euratom of 15 July 1980 amending the Directives laying down the basic safety standards for the health protection of the general public and workers against the dangers of ionising radiation. *Official Journal of the European Communities* L246, 17.9.80. p. 1.

Acknowledgement

The author wishes to thank colleagues for their help in preparing this paper. The views expressed, however, are those of the author and not necessarily the views of HSE.

LAW ON PROTECTION AGAINST IONISING RADIATION AND NUCLEAR SAFETY IN SLOVENIA

B. Breznik⁽¹⁾, M. Križman⁽²⁾, D. Škrk⁽³⁾, R. Tavzes⁽⁴⁾

⁽¹⁾Krško Nuclear Power Plant

⁽²⁾Slovenian Nuclear Safety Administration

⁽³⁾Health Inspectorate of Republic of Slovenia

⁽⁴⁾Ministry of Environment and Spatial Planning

Introduction

The existing legislation related to nuclear and radiation safety in Slovenia was introduced in 80s. The necessity for the new law is based on the new radiation safety standards (ICRP 60) and the intention of Slovenia to harmonise the legislation with the European Union. The harmonisation means adoption of the basic safety standards and other relevant directives and regulations of Euratom. The nuclear safety section of this law is based on the legally binding international conventions ratified by Slovenia. The general approach is similar to that of some members of Nuclear Energy Agency (OECD). The guidelines of the law were set by the Ministry of the Environment and Spatial Planning, Nuclear Safety Administration, and Ministry of Health.

The expert group of the Ministry of Environment and Spatial Planning and the Ministry of Health together with the representatives of the users of the ionising sources and representatives of the nuclear sector, prepared the draft of the subject law. The emphasis in this paper is given to main topics and solutions related to the control of the occupationally exposed workers, radiation safety, licensing, nuclear and waste safety, and radiation protection of people and patients.

General legislation

Nuclear legislation of the Republic of Slovenia has began twenty five years ago and covers the use of radiation sources and nuclear energy. The original features of this legislation were derived from international recommendations or regulations. For example, radiation protection standards were derived from ICRP recommendations and IAEA basic standards. Requirements for nuclear safety and liability of a nuclear operator were based on adopted international conventions.

Slovenia used to be a part of federal state of Yugoslavia and nuclear legislation was found in the Act on Radiation Protection and Safe Use of Nuclear Energy issued in 1984, its first version was from 1976. In late eighties ten executive regulations were put in place on radiation protection, four on nuclear installations, some more related to other activities and standardization, and a few decrees. Act on Liability for Nuclear Damage was issued in 1978. The former Republic of Slovenia issued also its own regulations on protection against ionizing radiation and nuclear safety and on nuclear liability. There were two agreements set between the former state and IAEA to obtain Agency assistance and to

implement safety standards for construction of research reactor in Ljubljana (1961) and nuclear power plant in Krško (1976).

The regulatory basis in new independent Republic of Slovenia has been provided by Act of Constitution in 1991, which specifies the regulations and agreements to be maintained from the former regulatory system. Some segments of the old legislation system were not coherent and excessively prescribing. The need for revision is due to adoption of the overall legislation changes, to the new radiation safety standards and international conventions, and due to approaching the European Union.

The validity of the new Law on Radiation and Nuclear Safety is scheduled for the year 2002. The draft of this Law covers: basic rules and conditions for radiation practices, protection of people against ionizing radiation, radiation and nuclear safety of the installations, radioactive waste, spent fuel, and transport, non-proliferation and physical protection, environmental and operational monitoring, intervention and remedial measures, expenses of source user and public expenses, compensation of nuclear installation due to limitation in use of the surrounding space, state report on radiation and nuclear safety, and penalties.

The new Law corresponds to previous acts introducing the Convention on nuclear safety (1996) and Joint convention on the safety of spent fuel and radioactive waste management (1999), and to other acts related to environmental protection, construction and mining, transport of dangerous goods, civil defence, and to internal affairs.

Guiding principles

The new Law is guided by the following main principles:

- implementation of all appropriate and reasonable measures of the state in executing this law to prevent a possible health detriment and radioactive contamination of the environment;
- justification of new methods or practices causing exposure to ionising radiation;
- optimisation of exposures of the practices to the level as low as reasonably achievable, economic and social factors being taken into account;
- the dose limits for total doses caused by all relevant practices and exposures due to natural sources;
- peaceful use of nuclear materials and technologies, according to international conventions;
- prime responsibility of source user or the operator for protection against radiation and for nuclear and radiation safety;
- liability of the source user and the operator;
- causer pays for remedial measures necessary to ensure radiation protection;
- emergency preparedness of the nuclear installation or radiation facility licensee;
- state's subsidiary measures in case of unknown or unclear polluters, and
- openness to the public.

Reporting to the authorities

The Law requires reporting to authorities by an application before start with the practice that can increase exposure of individuals, such as production, processing, use, storage, transport, export, import, disposal, other handling or possessing of radioactive materials. The practices which should also be reported are related to operation or maintenance of electrical equipment emitting ionizing radiation and containing components having voltage more than 5 kV, and other practices if defined by the government. The reporting is performed by notification to the authorities. There are exemptions foreseen in the reporting. In executive regulations, the government will define exemption levels, and also clearance levels of radioactivity to be used as basis for release from regulatory control. Sources of minor importance will only be registered.

The authorities are Ministry of Environment and Spatial Planning, and Ministry of Health in case of medical or veterinary practices. They maintain the registers of practices, nuclear materials and radioactive sources.

The radioactive sources are registered based on the notification to the competent authority.

In some cases the reporting is performed already by application for licence. Such exemptions are import, export or transit of radioactive materials or waste, multiple use of radioactive sources, and siting of the nuclear or radiation facilities.

Classification of installations

Classification of installations is based on environmental aspect and is related to site permit and construction licence procedure. The law defines nuclear installations, radiation facilities, and less important facilities related to a radiation practice.

Nuclear installations are nuclear power plant, critical or sub-critical reactor, research reactor, nuclear fuel fabrication, enrichment or reprocessing facility, spent fuel storage or repository, temporary storage or final repository of low or medium active waste, radioactive waste treatment or incineration facilities. Siting of nuclear installations correspond to the state's spatial planning rules and prescribed safety analysis of the location and report on to environmental impact.

Radiation facilities are defined as:

- radiation facility having a risk that the dose of the public may exceed the dose limit;
- facility with open radioactive source(s) having a risk of release to environment which may cause the dose exceeding the dose limit for individuals of the public;
- facility with the practice having releases to environment in a year in excess of ten times exempted activity levels;
- installation for mining, processing or enrichment of nuclear ore materials;
- depository of nuclear mineral ore or related hydrometallurgical tailings.

Less important radiation facilities are buildings with radioactive source(s), having a risk, that doses of individuals or workers inside them, exceed the dose limits.

Licensing

Authorisation, as defined by Directive 96/29/Euratom, is adopted in the Law as licence or permit. The licence for practice in case of nuclear installation or radiation facility is understood as operation licence.

Application for licence for new classes or types of practices or use of new radioactive source or new procedure should include report on justification of these practices.

The prior permit of Ministry for Environment is required for:

- operation and decommissioning of radiation or nuclear installations;
- the deliberate addition of radioactive substances in the production and manufacture of consumer goods and the import or export of such goods;
- the use of X-ray sets or radioactive sources and the use of accelerators with exception of electron microscopes, in other than medical or veterinary practices;
- disposal, processing or reuse of radioactive materials or contaminated materials, originating from the practice or usage of radioactive source if these materials are not previously authorized to be released from regulatory control;
- production or development equipment or means related to nuclear technology;
- import, export or transit of nuclear and radioactive materials;
- maintenance, calibration or similar work with radioactive sources, if not already included above.

Ministry of Health is competent to licence the use of radioactive sources or the practices in medical or veterinary field, as they are specified by the Directive 96/29/Euratom, and the control of, or other work with radioactive sources.

The licence for radioactive sources is based on the permit for the practice and written confirmation that the source is registered. In case of nuclear installations and radiation facilities the operating licence includes the licence for radioactive source or nuclear material.

The siting of nuclear installation shall be included in the general spatial plan of the state. For this, a special safety analysis of the site is required. In a case, that site permit is necessary to provide an agreement on conditions of environmental protection for nuclear or radiation facility, it includes radiation and nuclear safety requirements, defines contents of project documentation, levels of radiation in the environment, and limited use of the territory.

Location and construction permit for nuclear installations and radiation facilities correspond to Environmental Law and to Act on Buildings Construction.

For obtaining the construction licence, it is required to submit the Safety Report, specifications for safe operation, and the programmes for physical safety, organization, emergency planning, training, quality assurance, monitoring, test operation, radiation protection and assessment of protection of workers against radiation.

Permits are required also for start of operation or closure of nuclear installation, beginning or ending of decommissioning of nuclear installations or radiation facilities, closure of mining or

depository facilities of nuclear mineral or tailings. Before operation, the test operation permit is foreseen for new nuclear installations or radiation facilities. Application for operation licence should include updated Safety Report and the opinion of authorised technical organization.

For the radiation facilities with medical or veterinary practice, the procedure requires only practice licence and source licence, and report on assessment of protection of exposed workers, to permit the operation.

The permit for practice and the operation licence are issued for a fixed period of time. These licences can be changed by the Ministry in the case when:

- nuclear or radiation safety conditions are changed;
- there are reasons for protection of environment, human life or health in public interest;
- significant decrease of nuclear or radiation safety due to external or natural phenomena.

Radiation and nuclear safety

Operation of nuclear power plant is regulated on the basis of Convention on Nuclear Safety. The Law requires implementation of operating experience and corrective action programs, financial and human resources management, qualification of personnel, quality assurance, reporting, operational and environmental monitoring, emergency preparedness and evaluations of radiation and nuclear safety.

Periodic safety reviews for overall assessment of radiation or nuclear safety should be performed by the operators as defined by executive regulations.

A specialized radiation protection unit, authorized to perform radiation protection tasks and provide specific advice is required for nuclear installations and radiation facilities. For other practices, it is required to have a person responsible for radiation protection.

Changes or modification of the facilities and operating procedures should be classified according to the importance for nuclear or radiation safety. According to this the operator should evaluate and, as appropriate, notify the authorities, report, or should obtain the permit to perform the changes.

Ministry of Environment, as a result of their oversight activities, may stop the operation of nuclear installation or radiation facility in the following cases:

- if radiation or nuclear safety conditions are not fulfilled within reasonably defined time frame, required by the inspector;
- if the licensee does not provide, within due time, a revised assessment of radiation protection of the workers;
- if the licensee starts with maintenance activity or implements technical changes, important from a radiation or a nuclear safety aspect, without prior approval.

Revocation of licences for a practice is possible in the following cases:

- if radiation safety requirements, as a result of the oversight activities, are not fulfilled in a reasonable time frame;
- assessment of protection of exposed workers is not provided to the authorities or if the practice is conducted without approved assessment for more than six months.

Radioactive waste and spent fuel

It is required that the owner of radioactive waste or spent fuel should assure:

- storage and handling of the waste or spent fuel in accordance with current regulation;
- avoiding imposing burdens of waste or spent fuel removal on future generations as much as possible; and
- should minimise radioactive waste production.

Radioactive waste and spent fuel disposal is under responsibility of the state. The financing is assured by dedicated public fund. The costs for waste management rely on those who generate or possess the waste. The state is responsible to cover these expenses if the origin of the waste is not known.

The government establishes public agency to manage radioactive waste and spent fuel storage, treatment before removal, and final deposition. The producers may also temporary store or process the waste or spent fuel at the source, if they have the licence. Collection, transport and temporary storage of the waste before removal is a matter of a public service. The agency prepares national programmes for radioactive waste and spent fuel management.

Protection of individuals

General obligations related to the practice

The licensee should justify and prove that each new practice has benefits in relation to the health detriment, should justify already permitted practice in case of new important evidence about its benefit or detriment, should optimise the exposures as low as reasonably achievable, economic and social factors being taken into account, should use dose constraints in optimisation of the practice, should assure that doses of workers, apprentices, students and individuals from the population are within prescribed dose limits.

Protection of exposed workers

The undertaking should provide protection of exposed workers, students and practitioners by:

- prior evaluation of the risk and optimisation of radiation protection in all working conditions;

- classification of workplaces regarding the expected annual doses and to probability of potential exposures;
- classification of workers into categories;
- training;
- informing the personnel about technical, medical and administrative procedures related to all aspects of use of radioactive source;
- informing about the risk for health of workers and early declaration of pregnancy;
- monitoring of workplace and personal dosimetry;
- regular checking of emergency guidelines and effectiveness of protective equipment;
- regular calibration, checking and use of measuring instruments;
- medical surveillance of exposed workers.

These requirements should be fulfilled by the employer if there exist a possibility for the workers to exceed annual dose limits for members of the public.

The undertaking is responsible for arrangements for the radiological protection and to consult the qualified expert or the approved radiation protection service in case of assessment and implementation of these arrangements.

The Law explicitly defines the outside workers in accordance with the Directive 90/641/Euratom and gives the bases for the implementation of the Directive for these workers by controlling the outside undertakings as well as the operators.

In some cases the individuals are specifically protected: a worker who refuses specially authorised exposures exceeding the dose limits; a pregnant or lactating woman relocated to a new job, should be given an equal position; a person under 18 years of age should not become an exposed worker.

The Law has provisions to define responsibilities within the radiation practice, and to recognise approved medical practitioners, occupational health services, and qualified experts. It defines also basic rules for dosimetric services, central dose register, and dose information exchange.

Medical exposure

In accordance with the Directive 97/43/Euratom the improved concept of the medical exposure is elaborated. The medical practice exposure is regulated in a way requiring specific written protocols, programmes and assessments, and by introducing optimisation and evaluation of the exposures based on the exposures register information.

Exposure due to natural radiation sources

The competent authority should assure protection of individuals and monitoring of increased exposures due to natural radiation sources, or in case of some work activities. The protection of air crew and related actions should be required from the employer if there is a probability to exceed the dose limits.

Institutional framework and competent authorities

The Parliament approves state's spatial plan with siting of nuclear installations. The Minister of Environment and Spatial Planning may initiate this procedure.

The Government establishes advisory committees, and is competent to issue executive regulations or decrees, and to finance education programmes for authorized expert, development studies, independent expertises, and international collaboration in the field of radiation protection and nuclear safety.

The Ministry of Environment and Spatial Planning is competent authority in the field of nuclear and radiation safety, except medical or veterinary practices. The Ministry also approves qualified experts for radiation and nuclear safety and qualified technical organizations. The Ministry of Health is competent authority for protection of professionally exposed workers and individuals from the public, for medical and veterinary exposures, for the licencing of the related practices or radiation facilities with insignificant influence to the environment, for approval of dosimetric services, qualified experts or technical organizations, and for licences of radiation protection staff in the installations. Both ministries have dedicated units to perform licencing and administration procedures or to concur in some cases.

The two advisory bodies to advise the ministries, and to support decision of inspectorates or governmental administration are:

- Committee for topics related to radiation and nuclear safety, physical protection of nuclear materials, radioactivity in the environment and remedial measures after radiological incidents;
- Committee for topics related to protection of individuals against ionizing radiation, medical and veterinary use of radiation sources.

The Minister of Internal Affairs is competent for approval of physical protection plans and for non-proliferation control.

It is the responsibility of the Ministers of Environment, Health, Internal Affairs, and of Civil Defence to issue executive regulations or concurrence, as appropriate.

RADIATION PROTECTION REGULATIONS IN SLOVAKIA AND APPLICATION OF BSS AND EC COUNCIL DIRECTIVES PROVISIONS

D. Viktory

National Public Health Institute of the Slovak Republic, Slovakia

Slovakia, a part of the former Czechoslovak Federation, has been on the map of Europe as an independent country for only 10 years. It is a relatively small country with 5,5 million of inhabitants. The use of radiation sources in medicine, industry and research is extensive, but it corresponds with the size of country and the number of inhabitants. Nowadays there are nearly 600 licensee holders in medicine and in industry and 5 000 radiation sources in national registry.

The nuclear industry is relatively developed in Slovakia. Six nuclear reactors of type VVER 440 are in operation. One reactor is being decommissioned. There are also other nuclear installations in operation, such as interim storage for spent nuclear fuel in Bohunice, the conditioning center for radioactive waste in Bohunice, with incineration, solidification – bitumenation and cementation, vitrification unit, several experimental installations for solidification of radioactive waste operated by Research institutes for Nuclear Installations, and the near surface disposal for low and medium radioactive waste in Mochovce. It is expected that two reactors of NPP V1 in Bohunice will be shutdown in year 2006 and 2008, and that two units in Mochovce will be completed. The construction is interrupted now. The interim storage for spent fuel in Mochovce is in design stage and there is also a geological investigation for geological repository for high level radioactive waste and spent fuel. From this long list of important sources is it clear, that the legislation in radiation protection is and will be very important.

In the former Czechoslovakia, the responsibilities for nuclear safety and radiation protection were divided between two authorities. The federal authority for nuclear safety and regional authorities – regional hygienist for the radiation protection were responsible. Regional health protection authorities were managed by two separate Ministries of the Health of Czech and Slovak federal Republics. After the splitting of Czechoslovakia the development differs slightly in both countries. In the Czech Republic both authorities have merged, in Slovakia the people in power were not able to carry out such a radical change, so here the model of two independent authorities have remained. This system has been working since 1950s. After the splitting a small paradox has occurred. The newly established Slovak Nuclear Regulatory Authority has won the international reputation but in radiation protection you can perceive some prolonged scepticism and problems. One of the reasons are limited financial resources. I think, that in the health departments of all transformed countries there is a permanent lack of resources. This is also the reason for relative limited scientific background and support for radiation protection executive bodies. Despite of recommendations of some international missions and visits to join both authorities, there is no willingness in the Health Ministry to merge or to make an other basic change within the health department, as it was done in Lithuania. On the other hand, the ecological organizations mostly prefer to have two independent authorities.

National legal framework

There are two basic acts. Act no. 130/1998 on peaceful use of nuclear energy determines the authority, its responsibilities and basic requirements for nuclear safety. The authority is Nuclear safety regulatory, which is also empowered to issue particular regulations. Provisions on radiation protection are in the act No. 272/1994 on the public health protection. The legislation in radiation protection in Slovakia has been amended several times in last decade, not only because of radiation protection requirements. Even the system of authorities has been changed several times. At the beginning of nineties the authority was the state district physician, since 1995 it was chief hygienist of the Slovak Republic and since 2001 the authority for radiation protection has been the state regional hygienist for practices with standard medical and industrial radiation sources and the Chief hygienist for important practices (production of radionuclides, distribution of radiation sources) and important sources (nuclear installations, large accelerators). The act empowered the Ministry of Health to issue the regulation on radiation protection (Regulation No. 12/2001 on radiation protection). Ministry of Health is also empowered to issue instructions, standard procedures and guidances how to exercise the provisions of the act and regulation. At present some of them are in the phase of preparation.

There was a very good chance not only to apply the recommendations of BSS, but also to create completely new legislation system and regulations in connection with Slovak efforts to be accepted as a new member of EU. We did not succeeded completely in everything. In my opinion, one of the main reasons for this is that we do not have separate act for radiation protection. The provisions on radiation protection are only a part of the public health protection act. This leads to the problem which is not easy to solve, harmonization of sometimes different interests in the areas of epidemiology, communal hygiene, working hygiene, radiation protection and in other areas of preventive protection of health. In the crucial phase of the act and regulation development the possibilities for radiation protection experts to correct the provisions are limited. This can sometimes cause mistakes.

A brief summary of some important provisions on radiation protection in the act and regulation.

Terms which not fully correspond with BSS

- Radiation source, radioactive substance, source classes.

The radioactive substance is considered as the source only if exemptions levels are exceeded, and radioactive substance is considered as radioactive without any lower bound. Sources are divided into 6 classes. Sources of class one are not authorized, sources of class 2 and 3 are reported to the authority, and sources of class 4-6 are licensed.

- Supervised area is not defined and not used.
- Concerning exposed workers category, there is only category A and exposed worker without classification.

Basic framework and principles

Quantities and units, system of dose estimation, system of practices and interventions and basic principles are less or more defined in the sense of BSS.

Limits are the same as in BSS: effective dose 50 mSv in any particular calendar year, 100 mSv in any consecutive five calendar years period, and equivalent doses limits for lens of eye, skin and extremities.

Optimization principle has more or less the same wording as in BSS. It is necessary to optimize radiation protection measures if the expected annual doses in particular practice exceed the level of 1 mSv for workers, 10 mikroSv for the public, or annual collective effective dose of workers exceed 100 man mSv for workers, or 1 manSv for public, or collective effective dose of workers exceed 20man mSv per a particular task. There are also monetary values for collective doses in our regulation and additional conditions for the optimization process.

Authorities

Radiation protection authorities are part of the state administration structure. For licensing and radiation protection control for practices with radiation sources of classes 2-5 the regional hygienists are responsible. There are 8 regions in Slovakia. Regional hygienists are nominated by the Health Ministry and they are a part of the regional state administration.

The Ministry of Health, represented by the Chief Hygienist of the Slovak Republic is responsible for licensing of nuclear installations and some other specified important radiation sources.

The regional Institutes of Public Health act as the executive bodies in radiation protection supervision. The regulatory body or executive body for radiation protection supervision in NPPs is the National Public Health Institute of the Slovak Republic. There is a section for radiation protection and its part is the department for radiation protection in Nuclear Installations.

Licensing process

Practices with radiation sources of class 2 and 3 are notified (reported), practices with sources of class 4-6 are licensed. Any practice could be licensed if the applicant demonstrate that the provisions of legislation in radiation protection are met. In the Act, there is a list of documents which should be provided by the applicant. Some documents are approved by the authority, such as quality assurance programme, radiation protection programme, monitoring programme, emergency plan. The Nuclear Regulatory Authority and the Health Protection Authority issue a permit for the nuclear installations. Final licence is given by the local authority responsible for licensing construction and operation of installations, but the license can be issued only if both authorities issued the permits.

Empowerment of the regulatory bodies and supervision system

The act guarantees that the inspectors have the free entrance into the objects, buildings and controlled area. The inspectors have the right to ask for documents, results of monitoring, to take samples, to ask for the information and to look into the documents. But the rights of the inspectors are limited. In case he finds out some lacks or shortcomings during the inspection he does not have the right to stop the practice or to give sanctions on a place. In this case, an inspector has to inform the authority and to prepare a decision for the authority, which is empowered to give measures or sanctions, to stop the activity or practice or to cancel the license. These procedures, of course, can cause the delay of sanctions. My experience is that if any shortcomings appear the requirements of inspectors are most of time accepted and corrective measures taken.

Discharges, clearance of radioactive material

Release of radioactive substances into the air or surface water are licensed practices. The authority for health protection is responsible for license issuing. The Chief hygienist is responsible for that in nuclear installations. The dose constraints for planning and construction of the nuclear installations is 250 mikroSv for annual effective dose of the individual in most exposed reference group of the public. In licensing process the applicant has to propose the limits of discharges in the activity. He has to demonstrate that dose constraint is met and to assess the public doses due to discharges. The system of discharges processing should be optimized and should warrant that only effluents with activities are discharged if further processing is not effective and justified.

Clearance of radioactive substances from controlled area is also licensed. Generally the clearance of radioactive contaminated material could be allowed if the applicant demonstrate that clearance of material is optimal solution and that assessed annual individual public doses are bellow 10 mikro Sv and collective dose bellow 1 man Sv. The Regulation allows also higher public doses in special conditions if the radiation protection is optimized.

Radioactive waste

There is not a clear border in the responsibilities of the Nuclear Regulatory Authority and Health Ministry in case of treatment, conditioning and disposal of radioactive waste from NPPs. Only very close co-operation of both authorities could be the proper solution. We have to develop the co-operation system to avoid unnecessary overlaps of responsibilities.

Services

Activities which are important from the point of view of radiation protection (monitoring service, personal dosimetry, testing or examination of radiation sources, education in radiation protection) are licensed by the Health Ministry.

Emergency situation

The radiation protection authorities generally approve the emergency plans, but only Nuclear regulatory authority is responsible for nuclear installations emergency plans approval. The Regulation contains determined intervention and action levels according the BSS. Health protection authority is also empowered to order the intervention or protective measures and to license the remedial measures and management of residual activity.

Natural radiation

In the Regulation there are intervention levels for activities of natural radionuclides in drinking water, in building materials and radon in soil, in working places and in homes, and provision on protection of aircrew.

Conclusions

Generally the legislation in radiation protection in Slovakia harmonizes with BSS and EC directives. The system of radiation protection and the level which has been achieved is of a relatively good standard and is comparable with the level in developed countries. It is necessary to emphasize that the positive results have been achieved thanks to the cooperation within the IAEA, WANO and also ISOE. Mainly the persons responsible for radiation protection in NPPs have the wide international experience, they have many possibilities to compare and improve. We from regulatory body try also to apply our experience gained in IAEA projects and ISOE information net. The process of harmonizing of our legislation with BSS and EC directives is not direct and still not finished. There are some unexpected obstacles, sometimes we are disappointed with the results of our efforts. There are still some unnecessary inaccuracies and provisions, which should or could be better formulated and harmonized. We have to develop the third level of legislation – guidances and instructions. The existence of the two authorities should be carefully evaluated.

TRANSPPOSITION OF REQUIREMENTS SET OUT IN THE BASIC SAFETY STANDARDS FOR NUCLEAR FACILITIES IN LITHUANIA

G. Klevinskas

Radiation Protection Centre

A. Mastauskas

Radiation Protection Centre

Abstract

The legal basis for radiation protection allowing to protect people and the environment from the harmful effects of ionising radiation is established by the Law on Radiation Protection of the Republic of Lithuania (1999). The basic radiation protection requirements are described in [4]. The requirements set out by [4] related to occupational radiation protection of nuclear power plant workers are established and explained by [5], which sets out requirements for radiation protection of workers working at the nuclear power plant and for radiation protection of members of the public during the nuclear power plant operation. Radiation protection of outside undertakings is regulated by the [6]. Limitation of discharges of radionuclides into the environment from operation of nuclear facilities is regulated by [7]. The requirements of the above mentioned legal acts are in compliance with international standards and recommendations.

Introduction

Lithuania has one nuclear power plant – Ignalina NPP, which contains two RBMK-1500 reactors (actual thermal power output – 4 200 MW, electrical power capacity – 1 500 MW). The first Unit of Ignalina NPP went into operation at the end of 1983, the second Unit in August 1987. After Lithuania regained its independence in 1990, because of changed political thinking, when old standards have been replaced by national, constantly the “new” thinking is being implemented. This is also happened in the field of radiation protection. A number of new laws, Government Resolutions, regulatory documents (Hygiene Standards and Orders of Ministers) have been established which forms the legal basis for radiation protection in Lithuania. It shall be considered that one of tools which allows to keep the national radiation protection requirements in accordance with international standards, is the harmonisation of national legislation with the international requirements and recommendations.

The radiation protection requirements set out by the relevant Lithuanian radiation protection legislation directly related to nuclear facilities are presented in the paper.

Regulatory framework of radiation protection and responsibilities of the Radiation Protection Centre

The hierarchical structure of the Lithuanian legislation related to radiation protection is presented in the Figure 1.

Figure 1. The hierarchical structure of the Lithuanian legislation related to radiation protection



The list of main regulatory documents in force on nuclear safety, radiation protection and radioactive waste management related to nuclear facilities operation in Lithuania includes following laws, basic regulations and standards:

- Law on Radiation Protection (No. VIII-1019, 1999).
- Law on Nuclear Energy (No I-1613, 1996, as amended 1999).
- Law on the Management of Radioactive Waste (No. VIII-1190, 1999).
- Law on Environmental Protection (No. I-1352, 1992, No. I-2223, 1996, 1999).
- Hygiene Standard HN 73:2001 “Basic Standards of Radiation Protection” (2001).
- Hygiene Standard HN 87:2001 “Radiation Protection in Nuclear Power Plant” (2001).
- Hygiene Standard HN 83:1998 “Radiation Protection and Safety of Outside Workers” (1998).
- Normative Document LAND 42 – 2001 “Limitation of Radioactive Discharges from Nuclear Facilities, Permitting of Discharges and Radiological Monitoring” (2001) and other legal acts.

The main law establishing the legal basis for radiation protection allowing to protect people and the environment from the harmful effects of ionising radiation is the Law on Radiation Protection No. VIII-1019 adopted on 12 January 1999 [3]. The Law regulates relations of legal persons, enterprises without the status of a legal person, and natural persons arising from activities involving sources of ionising radiation and radioactive waste management.

According to Article 3 of [3], all practices shall be authorised and conducted in accordance with the following basic principles of radiation protection:

- [1] The principle of justification of the operation of sources of ionising radiation – the economic, social and other benefits yielded by all types of practices involving operation of sources of ionising radiation to individuals or society must outweigh the detriment radiation causes to human health and the environment.
- [2] The principle of optimisation – any kind of exposure of individuals and society must be as low as reasonably achievable, economic and social factors being taken into account.
- [3] The principle of limitation – the sum total of doses resulting from all types of practices may not exceed the fixed value, with the exception of a personal dose received by a patient for his own health care and an individual (other than as part of his occupation) voluntarily helping a patient or participating in medical and biomedical research.

According to the Article 7 of the [3], the Radiation Protection Centre is a body co-ordinating the activities of executive and other bodies of public administration and local government in the field of radiation protection, exercising state supervision and control of radiation protection, monitoring and expert examination of public exposure. As regards the nuclear facilities, one of the main aims of the Radiation Protection Centre is to supervise, control and to demand that the activities carried out at nuclear power plant are in compliance with radiation protection requirements, established by [3], [4], [5], [6] and other legal acts.

Requirements for Occupational Radiation Protection in Nuclear Power Plant

The main regulation which sets out requirements for occupational radiation protection of workers working at the nuclear power plant and for radiation protection of members of the public during the nuclear power plant operation, is the Lithuanian Hygiene Standard HN 87:2001 “Radiation Protection in Nuclear Power Plant” [5]. It was approved by the Order of the Minister of Health and came into force on 1 April 2001. The Hygiene Standard shall be applied for all legal and natural persons conducting their activities at the NPP. The requirements of the Hygiene Standard are in compliance with the requirements and recommendations established by [1], [2].

According to Article 10 of [3], where the responsibilities of a licensed legal person are described, the main responsibilities for ensuring the radiation protection of nuclear power plant workers are delegated to the license holder.

As regards the limitation of occupational exposure, the license holder shall ensure that doses of nuclear power plant workers do not exceed the dose limits established by [4], excluding exceptional circumstances, [4] establishes dose limits for occupational and public exposure which are in compliance with [1] and are presented in Table 1.

The [5] requires the license holder to establish the investigation levels and dose constraints for nuclear power plant workers. The investigation levels shall be established in order to fix the achieved real level of exposure and to ensure optimal measures for protection of workers against the dangers of sources, used during the nuclear power plant operation. Investigation levels shall be regularly reviewed taking into account the radiation protection conditions at the nuclear power plant.

Table 1. Dose limits for occupational and public exposure

Application	Dose limit	
	Occupational	Public
Effective dose	100 mSv in a consecutive 5 year period, subject to a maximum effective dose of 50 mSv in any single year	1 mSv in a year, in special circumstances – up to 5 mSv in a single year provided that the average dose over 5 consecutive years does not exceed 1 mSv per year
Annual equivalent dose		
In the lens of the eye	150 mSv	15 mSv
For the skin	500 mSv	50 mSv
For the extremities (hands and feet)	500 mSv	

As regards the limitation of public exposure that might cause the nuclear power plant operation, [5] establishes the dose constraints for the members of public. The annual effective dose constraint for the members of public because of operation of nuclear power plant is 0.2 mSv. The annual dose constraint is used by [7] for setting the maximum permitted levels of discharges.

According to requirements set out in the [5], the radiation protection programme shall be established in the nuclear power plant, where a set of measures shall be implemented in order to protect workers from the negative impact that may cause the ionizing radiation. Following items shall be included in the programme:

- 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 rules for their application;
- main premises, control systems for assurance of radiation protection;
- requirements for management of radioactive waste;
- radiation protection measures applied during the accident;
- application of optimisation principle (ALARA) and measures on exposure reduction;
- programs of health surveillance;
- mandatory training of workers and their instructions.

The establishment of a comprehensive radiation protection programme is also recommended by the [1].

The license holder shall ensure that all management procedures assigned for the implementation of the radiation protection programme at NPP, are performed in accordance with the requirements of the quality assurance programme.

According to requirements of [5], the premises of the NPP shall be divided into controlled and supervised areas. Controlled area is an area subject to special rules for the purpose of protection against ionising radiation or of preventing the spread of radioactive contamination and to which access is controlled. The supervised area is defined as an area subject to appropriate supervision for the purpose of protection against ionising radiation. Depending on the dose rate, surface and air contamination levels, the premises of the controlled area are divided into three categories (Table 2). The license holder shall delineate the boundaries of controlled area by physical and other means, acceptable from the point of view of radiation protection, post warning symbols, appropriate instructions shall be at the entrances and in other appropriate locations within the controlled area. The access to controlled area is restricted by means of physical barriers and is organized through sanitary cleaning points.

Table 2. Classification of premises of controlled area of NPP

Category of premises		I	II	III
Controlled values	Dose rate, $\mu\text{Sv/h}$	>56	12-56	<12
	α contamination of surface, $\text{Bq}\cdot\text{cm}^{-2}$	>20	4-20	<4
	β contamination of surface, $\text{Bq}\cdot\text{cm}^{-2}$	>266	40-266	<40
	Concentration of airborne activity, $\text{Bq}\cdot\text{m}^{-3}$	>1110	185-1110	<185

All procedures within the controlled area, which may cause increased exposure of workers, shall be determined by the license holder and shall be performed by using the appointments-permits and assignments. The license holder is required to make the list of works, which shall be performed according to appointments-permits and assignments.

The license holder is responsible for organisation, implementation, carrying out and improvement of workplace monitoring and of individual monitoring of every worker, working within controlled area. The license holder is responsible for preparation of the program for individual monitoring and monitoring of workplaces. The programme shall be agreed with the Radiation Protection Centre. Taking into account the changed working conditions, the license holder shall regularly review and, if necessary, renew the programme of workplace and individual monitoring.

All workers working within controlled area, depending on radiation situation in workplaces, shall be provided with individual protective equipment.

The nuclear power plant workers and outside workers shall be regularly trained and instructed in the field of radiation protection and safety. The frequency of training and instructing is established by the Order No. 171 of the Minister of Health “On Procedure of Mandatory Training and Instructing for Persons, Responsible for Radiation Protection and Workers, whose Work Involves Sources of Ionizing Radiation” (1999). 240, 60 and 30 hours training 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. The training programs shall be agreed with the Radiation Protection Centre.

The medical examination of workers shall be carried out once per year before the starting of activities with sources of ionizing radiation and during the work according to the requirements of the Order No. 301 of the Minister of Health “On Prophylactic Medical Examinations at the Institutions of Health Care” (2000). Depending on the contra-indications detected, activities within the controlled area are either forbidden or limited.

The [5] also establishes requirements for optimization of radiation protection. The establishment and implementation of the ALARA programme is required. The main aim of the programme is to ensure that the exposure of workers is being kept as low as reasonably achievable, social and economical factors taking into account. There is the ALARA group established at the plant. The main tasks of the group are to carry out comprehensive analyses and prepare proposals for exposure reductions, prepare the dose budget for the following year, plan the dose commitments for outages, prepare reports of implementation of ALARA programme, etc.

Following conditions are subject to successful implementation of ALARA programme:

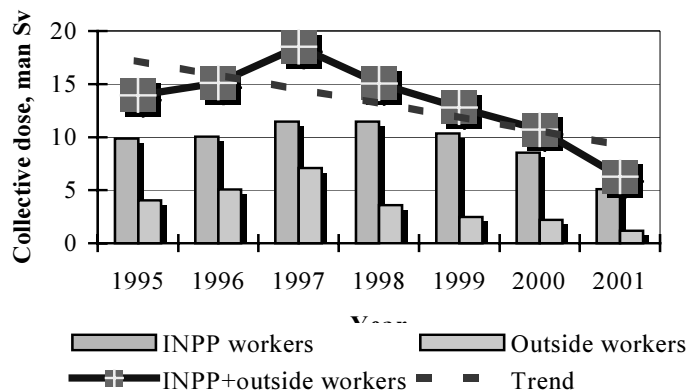
- proper work organisation;
- improvement of working conditions;
- perfection of technological processes;
- training of personnel;
- implementation of quality assurance programme;
- improvement of safety culture;
- evaluation of influence of “human factor”.

Decontamination of systems and components are carried out before starting the activities that may lead to increased exposure, e.g., the decontamination of the main circulation circuit is one of methods for the reduction of doses. The activities, during which higher exposure is expected to be received, are carried out by application of following means: installation of lead blankets, application of distance equipment, video-control systems etc. The successful implementation of the ALARA principle is reflected in the workers exposure results.

The occupational exposure results of Ignalina NPP workers and outside workers during the period from 1995 to 2001 are presented in Figure 2.

From 1995 the occupational exposure results of NPP workers and outside workers have decreasing inclination. During the period 1995-2001 the collective dose reduces in average of 1.33 man Sv each year.

Figure 2. Occupational exposure results of Ignalina NPP and outside workers during the period 1995-2001



Conclusion

The requirements of Lithuanian radiation protection legislation directly related to nuclear facilities are in compliance with international requirements and recommendations.

References

- [1] International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources. Safety Series No. 115, IAEA, Vienna, 1996.
- [2] European 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.
- [3] Law on Radiation Protection of the Republic of Lithuania, Vilnius (1999).
- [4] Lithuanian Hygiene Standard HN 73:2001 “Basic Standards of Radiation Protection”, Vilnius (2001).
- [5] Lithuanian Hygiene Standard HN 87:2001 “Radiation Protection in Nuclear Power Plant”, Vilnius (2001).
- [6] Lithuanian Hygiene Standard HN 83:1998 “Radiation Protection of Outside Workers”, Vilnius (1998).
- [7] Lithuanian Normative Document LAND 42 – 2001 “Limitation of Radioactive Discharges from Nuclear Facilities, Permitting of Discharges and Radiological Monitoring” (2001).

**FRENCH GOVERNMENT REGULATION – NEW ENACTMENT ON OPTIMISED
RADIATION PROTECTION OF WORKERS EXPOSED TO IONIZING RADIATION –
IMPLEMENTATION AT A FRENCH NUCLEAR POWER PLANT
BELONGING TO THE EDF GROUP**

J. Lebeau

EDF Nuclear Power Plant Operations Division

Optimisation of radiation protection, a binding substantial commitment

The radiation protection system defined by ICRP 60 and included in European Directive no. 96/29 is based on the three general principles of radiation protection : the justification of practices, the optimisation of radiation protection and the limitation of individual exposures.

Since 1988, French law requires for nuclear power plant an optimisation approach (ALARA) in the case of maintenance activities (equipment, methods and work organisation) relying on a specific structure: the department with special responsibility for radiation protection.

Recently, in December 1998, this general principle of optimisation was enhanced by the obligation to take a predictive approach in order to implement it. For all operations carried out in Radiation Controlled Areas (RCA), the law requires a prior estimation of individual and collective doses to which workers might be exposed, followed by the measurement and analysis of radiation doses actually absorbed during the activity.

Implementation at an EDF nuclear power station: Tricastin

With a view to consistent implementation on all its plants, EDF has drawn up a set of specific reference guidelines. These guidelines define the activity and describe a procedure for carrying out individual and collective dose forecasts for each activity. They also include a set of activity grading criteria based on the activity's radiological risk factor, a procedure for dealing with each level of risk, and a procedure for monitoring operational radiation exposure.

They deploy the three phases of the ALARA approach: planning, implementation and experience feedback.

Definition of the activity

Nuclear power plant operators are entrusted with maintenance, monitoring or operational activities. These activities vary according to time and place. This is why EDF defines an RCA activity either as being structured by a procedure (e.g. maintenance) or as a non-structured activity (e.g. logistics, service facilities or operations).

Dose forecasting

In the case of structured activities, projected dose forecasts are conducted on the basis of a procedure (break-down into basic phases, exposure times, number of workers involved, etc.), expected radiological conditions (RP surveys) and any available experience feedback.

Dose forecasts for periodic activities are based on experience feedback. As of April 2002, dose forecasts for periodic activities carried out at Tricastin NPP will be conducted on a target basis of 0.01 mSv per hour spent in the RCA.

Radiological risk levels

Implementation of this optimised approach is adjusted based on the radiological risk factor. However, what criteria are applied? Collective dose level, individual dose level, radiological environment, recurrent nature of the activity, exposure time, staff numbers concerned. At Tricastin NPP, three criteria are applied: collective dose, individual dose, and the activity's equivalent dose rate. Values for each of these criteria are determined by the site.

Example:

- Level-0 (non-rated) risk factor for a collective dose of < 1 man-mSv or an equivalent dose rate of < 0.1 mSv/h.
- Level-1 risk factor for a collective dose of 1 to 10 man-mSv or an equivalent dose rate of < 2mSv/h.
- Level-3 risk factor for a collective dose of > 30 man-mSv or an equivalent dose rate of > 40 mSv/h.

Activities performed during the last overhaul accounted for 1 300 man-mSv. Of these activities, 5% were level 3, 3.11% were level 2 and 51% were level 1.

Optimisation analysis: Devising actions liable to reduce exposure levels, while continuing to carry out "reasonable" actions

Optimisation analyses are adjusted according to the radiological risk factor. They are designed to identify elements contributing to dose (sources, work conditions: ergonomics, tools, handling, lighting, additional protective clothing, fallback area, radiological cleanliness, scheduling, etc.), as well as the means of reducing dose (technique, organisation, shields, option performance assessment, classification, choice of options). They are conducted on the basis of gradually itemised checklists, either by the craft involved in the case of level 1 activities, or by the RP department in the case of level 3 activities. Once the analysis is completed, a dose target (individual and collective) is set and included in the work package distributed to the workers, including the chosen options. At Tricastin NPP, this phase will be implemented progressively for level 3 activities during the next outage (April 2002).

Dose monitoring and experience feedback analysis

During the activity, received individual doses are measured and recorded manually. Collective dose is calculated and compared with the target. Deviations from radiological or technical conditions liable to modify the dose target are identified. Once a deviation exceeds a threshold set by the plant, the RP department is alerted in order to implement appropriate actions, together with the craft concerned. Tricastin NPP thresholds are 5 man·mSv (collective dose) or 0.5mSv/h (equivalent dose rate).

Upon completion of the activity, expected individual and collective doses are compared with actual figures. In the event of significant discrepancies, depending on the individual and collective dose thresholds set by the site, an RP experience feedback analysis is requested.

Tricastin NPP has set the threshold at $\pm 20\%$ and 2 man·mSv (collective dose), or at $\pm 50\%$ and 0.5 mSv (individual dose).

Difficulties in implementation

Whenever activities are assigned to a contractor, the latter must be involved at every stage. Dose forecasting practices have been standardised.

End-of-job analyses have shown that RP survey measurements are sometimes taken at a fair distance from the work-site: a reference distance has been established.

The computerised radiological data collection program is not suited to operational dose monitoring (different zoning, identification at RCA entrance). Monitoring is performed via the manual collection of individual dose data.

SESSION 2

QUAD CITIES UNIT ONE COBALT-60 EXPERIENCE

K. Ohr, E. Anderson, D Barker

Exelon Generation Company, Quad Cities Generating Station, Cordova

Executive summary

In October 2000, Quad Cities Generating Station entered its 16th Refueling outage on Unit 1 and discovered dramatically increased radiological conditions in the Drywell as well as Turbine Building steam-affected areas. The root cause behind the unexpected radiological conditions was determined to be a reaction between recently-injected Noble Metals (chemical compounds containing Platinum and Rhodium) and Depleted Zinc Oxide passively injected through the Feedwater system. This reaction, coupled with the historically high inventory of Cobalt-60 on in-core components, resulted in a loosely-bound corrosion layer on the fuel that was readily transported rather than the desired tightly-bound layer anticipated from Zinc (DZO) injection.

This excursion was previously unidentified in the industry as the practice of Noble Metals Chemical Addition was new to the domestic-Boiling Water Reactor (BWR) industry and has resulted in industry guidance on both Zinc Injection and Noble Metals Chemical Addition. Extent of condition of this issue has extended beyond Quad Cities Unit 1 and includes, to a lesser extent, Quad Cities Unit 2 and several other US BWRs.

Background

As an older Boiling Water-type Reactor (BWR), Quad Cities management has been engaged in evaluating chemical remedies to both mitigate Inter-Granular Stress Corrosion Cracking (IGSCC), and reduce Source Term creation and transport. This quest for Optimal Water Chemistry has included Hydrogen Addition (HWC), Depleted Zinc Oxide (DZO) Injection, and Noble Metals Chemical Addition (NMCA). The implementation dates are shown in Figure 1.

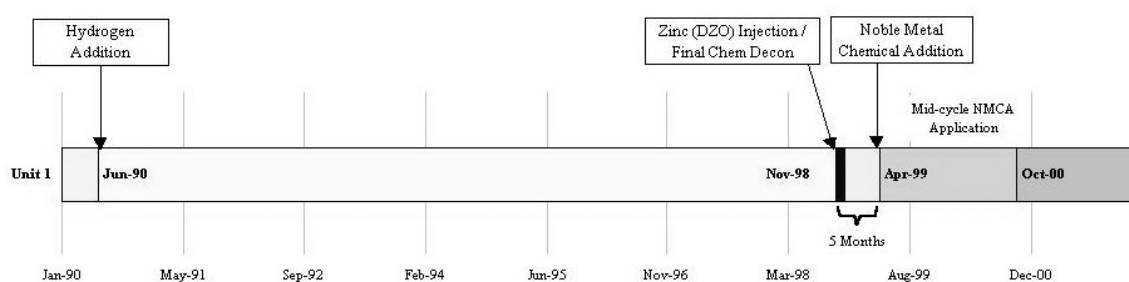
The primary purpose of HWC is the protection of stainless steel components by scavenging oxygen, ultimately decreasing corrosion on the piping surfaces. Several negative side-effects exist in conjunction with HWC, namely the increase in N-16 production and resultant high energy gamma dose rates during power operation and the costs associated with the high injection rates needed to maintain the required hydrogen concentration throughout the entire volume of water passing through Recirculation piping. At Quad Cities, HWC effectively protects core internals from IGSCC but only marginally protects Recirculation Piping while resulting in increases in dose rates in Steam-Affected areas five to seven times.

Depleted Zinc Oxide (DZO) is injected to reduce the amount of ⁶⁰Co incorporated into the primary system corrosion films and thereby reduce dose rates on primary system piping. The Zinc competes with cobalt for sites in corrosion films and inhibits corrosion on stainless steel surfaces. DZO also suppresses the release of established ⁶⁰Co from fuel cladding and in-core cobalt-bearing

materials. Implementation of DZO injection at Quad Cities Unit 2 was effective in reducing build-up of dose rates on Recirculation Piping and resulted in the lowest dose outage in Quad Cities history at 149 rem (1.49 Sv) in February, 2000.

Noble Metals Chemical Addition (NMCA) consists of Platinum and Rhodium injected into the primary system where they deposit on the piping and other vessel surfaces. Recirculation System piping and vessel internals are further protected from inter-granular stress corrosion cracking (IGSCC) and less hydrogen (HWC) is required to be injected due to the catalytic effect of Noble Metals on the piping surfaces. In terms of HWC consumption, NMCA is highly effective and reduces the required Hydrogen injection flow rates to one-fifth of former flow rates.

Figure 1. Noble metal chemical addition timeline



Each of these components affects the oxide layer on the fuel and other metal surfaces in and outside the core. Changes to any of these components affect both the composition of the oxide layer, and how tightly or loosely it is bound to the metal surfaces. This, in turn, affects the concentration of various isotopes (namely ^{60}Co) in reactor water.

Unit 1 shutdown radiological conditions

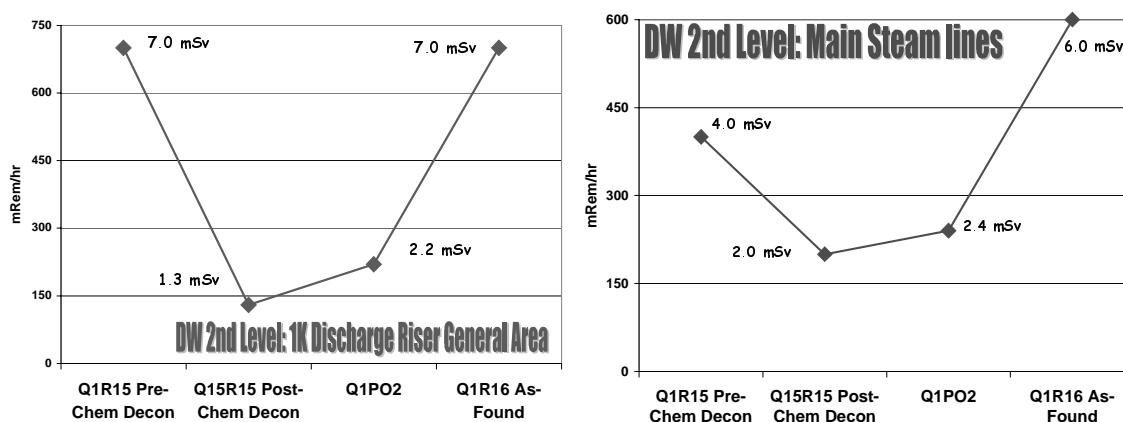
Unit 1 commenced its sixteenth Refueling outage (Q1R16) on 14 October 2000 with radiological conditions in the Drywell expected to mirror those of Unit 2 seen during its fifteenth Refueling outage (Q2R15) on February 2000. The radiological conditions on Unit 2 reflected the best Drywell conditions seen without the use of Chemical Decontamination of Recirculation piping in plant history. Injection of DZO on Unit 1 was planned to result in equally beneficial conditions during the Q1R16 outage.

The actual As-Found radiological conditions included the following anomalies:

- Reactor Water Chemistry spikes in activity at shutdown including increases in ^{60}Co activity approximately 15 times normal eventually increasing by a factor of 1 000 early in the outage.
- Drywell dose rates elevated by three to five times expected values (similar to what was historically seen pre-Chemical Decon in past outages).
- Secondary (Steam Side) dose rates elevated two to five times normal with the Moisture Separator elevated ten times normal.

The most dramatic impact was seen on the second elevation of the Drywell where the largest scope of inspection and maintenance activities were scheduled to be performed (Figure 2).

Figure 2. Quad cities dose rate changes in Drywell



Radiological response

The increased dose rates identified did not match any known model at that time. This led to several major actions to allow both for exposure reduction and for the need to complete the required Refuel outage activities.

- Installation of additional lead shielding (including shielding of the Main Steam Lines – now a significant source).
- Deferral of high dose work scope where prudent.
- Re-evaluation of all ALARA planning packages and respiratory requirement evaluations.
- Increased radiological job coverage with augmented technician and management staffing utilising resources from other Exelon nuclear stations.
- Implementation of daily Station ALARA Committee meetings.

In parallel with outage exposure control activities, an expert team was formed to determine the root cause of the unexpectedly high dose rates. This team was comprised of personnel from the Quad Cities Generating Station, other Exelon Stations and Corporate RP/Chemistry experts, General Electric, and Electric Power Research Institute (EPRI). Three additional teams were formed to begin evaluation of other long-term consequences of the conditions and likely remedies.

Root cause

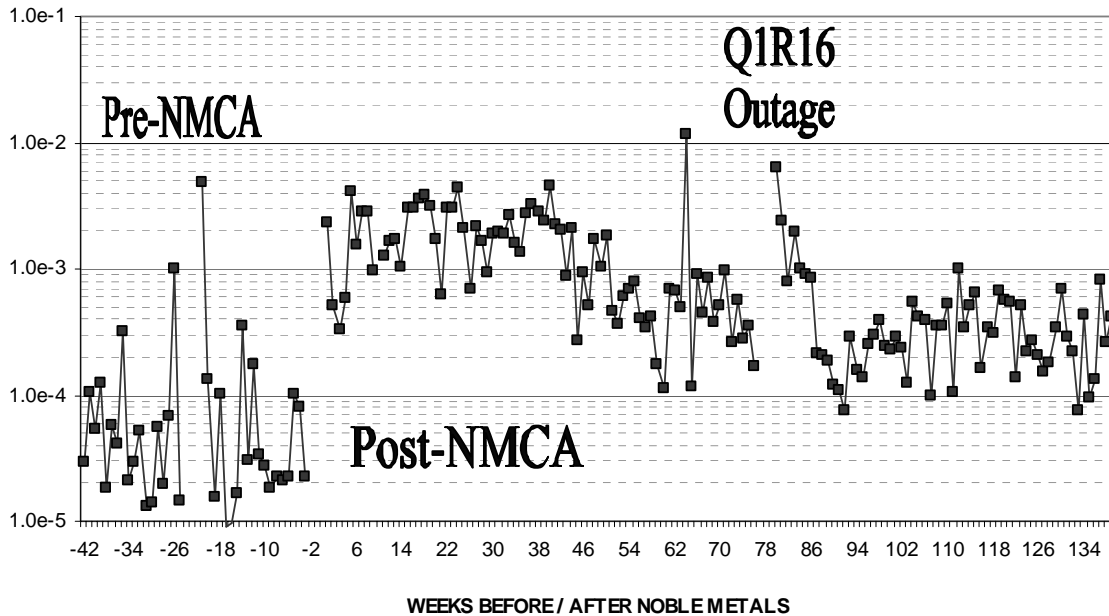
The root cause of the Q1R16 high shutdown Drywell dose rates was determined to be a combination of the fuel crud corrosion layer not being optimally stabilised and the high initial ^{60}Co inventory in the primary coolant. Other contributing causes were determined to be application of NMCA during a mid-cycle outage (with no immediate fuel removal) and excessive Hydrogen (HWC) cycling.

Simply stated, historically high ^{60}Co levels at Quad Cities Unit 1 were disturbed when NMCA was implemented on a system where DZO was not applied for a long enough interval to allow the fuel deposit DZO-affected corrosion films to stabilise. This condition was further exacerbated when DZO was not injected at high enough concentrations to effectively stabilise the fuel deposits.

Chemistry parameters

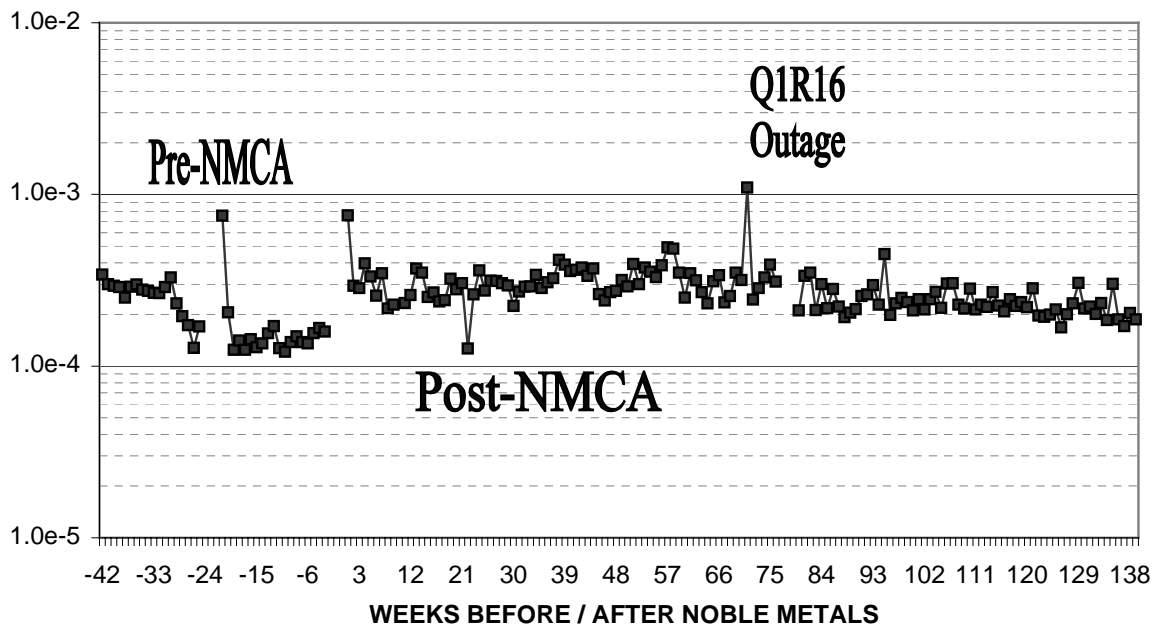
In retrospect, several trends become readily apparent. Insoluble ^{60}Co concentrations in reactor water increased by a factor of fifty after the application of NMCA as shown below in Figure 3.

Figure 3. Insoluble ^{60}Co concentrations in reactor water



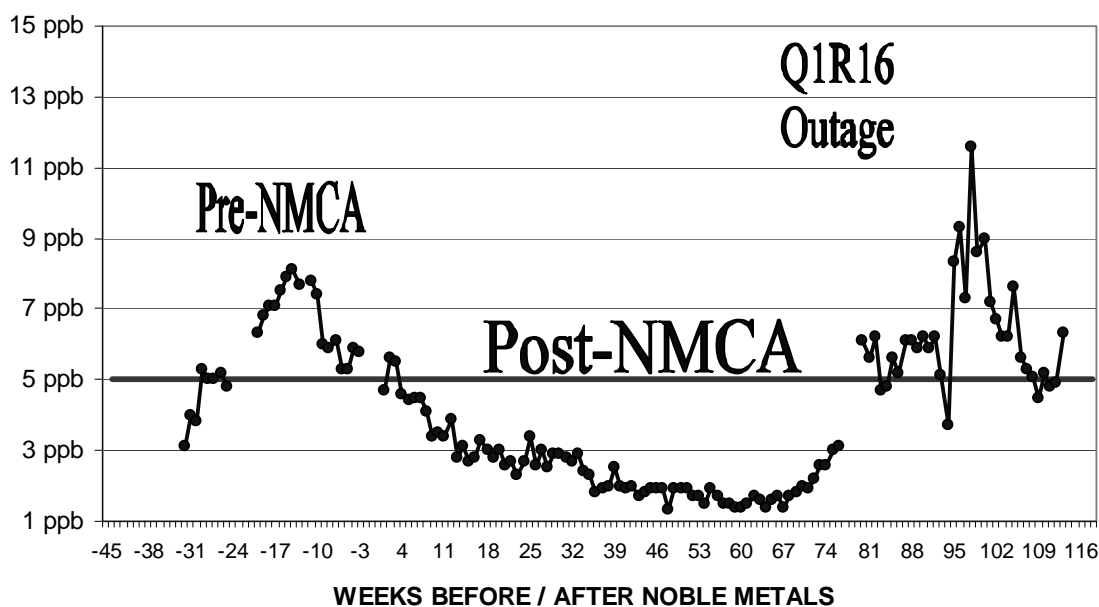
Similarly, soluble ^{60}Co concentrations were also seen to increase by as much as a factor of two in Figure 4.

Figure 4. Soluble ^{60}Co concentration in reactor water



Finally, DZO concentration in reactor water steadily decreased to 2 ppb despite no significant change to the input rate (Figure 5).

Figure 5. Depleted zinc oxide concentrations in reactor water



These indications were noted throughout the cycle leading up to Q1R16, but were dismissed through discussions with the vendor and the belief that the increased ⁶⁰Co concentrations were actually a positive indication of the preferential removal of ⁶⁰Co from the piping that would result in final removal of the ⁶⁰Co via the Reactor Water Clean-Up (RWCU) system. The lack of an independent confirming indicator (i.e. actual dose rates) due to a 510-day continuous run on the unit also prevented early identification and actions.

Lessons learned

Many technical and management lessons have been learned as a result of the Unit 1 high dose rates. Initially, the Quad Cities experience was communicated to the industry with recommendations by General Electric as part of Service Information Letter (SIL) 631 (subsequently revised once). This letter documented their initial position regarding DZO injection rates (5-10 ppb) and other chemistry parameters.

Beyond the technical recommendations, this event also served to reinforce the need for open communication between departments and to differentiate between laboratory (test) results and real-world results. When discrepancies are noted between hypothesized responses and actual results, actions must be developed and the differences investigated immediately.

Extent of condition

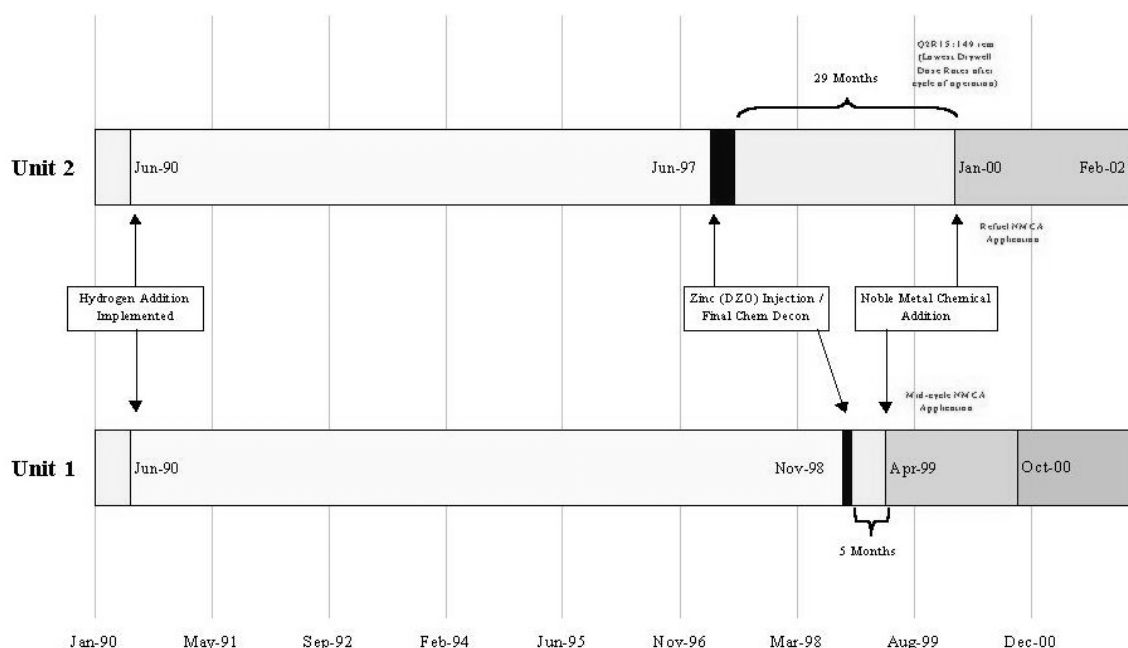
The anomalies noted on Unit 1 represent a latent event due to the time lapse between the triggering (initiating) event and the actual identification of the problem. The time between Quad Cities

implementation of NMCA and outage confirming dose rate anomalies was 510 days of continuous run on Unit 1. During this time, twelve other NMCA applications were performed in the United States. The issue of increased source term transport has now been identified on several other units (Nine Mile Point Unit 1, LaSalle County Station Unit 1, Peach Bottom Units 1 and 2, and Quad Cities Unit 2).

Unit 2 increases were mitigated by several differences in the various chemistry implementations, including:

- A 29-month DZO injection period prior to NMCA (versus 5 months on Unit One as shown in Figure 6).
- NMCA application at End-of-Cycle versus Mid-Cycle.
- Mid-course correction in DZO concentration on Unit Two (39 weeks below 5 ppb goal versus 73 weeks).
- Better understanding of ^{60}Co chemistry and impact.

Figure 6. Summary of Quad Cities Chemical Treatment Program for Units 1 & 2



Increases were noted in the same locations as Unit 1 but to a lesser magnitude and were mitigated through planned contingency shielding packages in all areas except the Drywell under-vessel area and the Refuel Floor Cavity post-draindown for reassembly. Even with mitigation of most of the impact, areas such as the Recirculation Piping discharge risers still doubled (2x) in dose rates.

With the elevated ^{60}Co now identified on both Units, long-term planning and evaluation is underway on the proper recovery measures. These will include removal of remaining Stellite (cobalt)-bearing Control Rod Blades, Chemical Decontamination of Suction and Discharge Recirculation Piping, installation of the Steam Dryer Modification to reduce moisture carry-over, and installation of permanent Drywell Shielding.

**IMPACT OF MAIN RADIOLOGICAL POLLUTANTS ON CONTAMINATION RISKS
(ALARA) OPTIMISATION OF PHYSICO-CHEMICAL ENVIRONMENT
AND RETENTION TECHNIQUES DURING OPERATION AND SHUTDOWN**

M. Berger

Électricité de France/SEPTEN

J.L. Bretelle

Électricité de France/GDL

A. Rocher

Électricité de France/GPR

1. Introduction

The goal of this paper is to precise the behaviour of different radiochemical species in the primary coolant of PWR plants. Managing these pollutants must lead to limit Reactor Coolant System (RCS) walls “over-contamination” to decrease the dose rates during the maintenance operations (ALARA).

In French Plants, ^{60}Co , silver and antimony represent the major radiochemical pollutants which require a good knowledge of the different phenomena to ensure the lowest contamination risks.

The stakes deal with the control and the optimisation of collective and individual doses including waste treatment with low costs. These stakes represent primordial elements of nuclear acceptability.

2. Normal contamination of circuits

In French PWR plants:

- More than 90% of integrated doses are due to contaminated walls in contact with the primary coolant because of activated corrosion products.
- 80% of total dosimetry are integrated during outages.

In the absence of specific pollution, the mean contribution of out of core deposits to dose rates is as follows (results expressed as percentage equivalent dose rates in Sv/h):

- $^{60}\text{Co} = 50\%$ (the older the plant, the greater the ^{60}Co quantity);
- $^{58}\text{Co} = 30\%$ (with a higher level in 1300 MW units);
- $^{110\text{m}}\text{Ag} + ^{124}\text{Sb} + ^{51}\text{Cr} + \dots = 10\%$.

Usual contaminants come mostly from materials/coolant interactions leading to soluble, particulate, colloidal products transport and corrosion products activation (table 1).

Table 1. **Gamma and beta energies and emission percentage (> 50%) of major pollutant nuclides**

Nuclides / Half-life	Gamma energy keV (% emission)	Beta energy keV (% emission)
⁵⁸ Co / 71 days	811 (100%)	
⁶⁰ Co / 5.3 years	1 173 (100%)	318 (100%)
	1 333 (100%)	
^{110m} Ag/ 253 d	658 (100%)	86 (55%)
	885 (80%)	
¹²⁴ Sb/ 60 d	603 (98%)	610 (51%)
¹²² Sb/ 2.7 d	564 (66%)	1 400 (63%)

The dose rates can increase in case of more important corrosion products transport [1] or of incidental pollution. In the latter case, the ratio of species contributing to dose rates is modified.

3. Incidental over-contamination

The major pollutants responsible for over-contamination in EDF plants are:

- ⁶⁰Co coming from stellite degradation;
- ^{110m}Ag coming from control rod perforation and probably from materials like seals;
- ¹²²Sb / ¹²⁴Sb coming from pumps of auxiliary system.

Given the importance of over-contamination with respect to dosimetry, representing 10 to 30% of outages dosimetry in the concerned units, a specific program is being set up in France, in the context of the ALARA project. The goal is to optimise diagnosis, surveillance, prevention and remedies to reduce pollutions even though priority is set on prevention.

3.1 Cobalt 60 contamination – “hot spots”

3.1.1 Hot spots sources

In most cases, hot spots are due to particles of cobalt activated by a neutron flux (⁶⁰Co) mainly from hard facing surfaces equipments (stellite, rich in cobalt) in the RCS (valves, pumps, internals, etc.).

3.1.2 Impact on dosimetry

The contribution of hot spots to shutdown dosimetry may appear to be marginal in French PWR reactors (2 to 4%), but becomes more significant (15 to 25%) for the units affected. This excess

dosimetry has to be taken into account, particularly for the most exposed workers. Approximately ten French PWR units have been affected by this phenomenon over the last 15 years [2].

3.1.3 *Hot spots behaviour – Indicators*

Surveillance is designed to inform the site as early as possible, of the presence of hot spots (mapping) in order to take the appropriate measures to prevent their propagation and/or to eradicate them.

During unit operation, most hot spots will remain fixed to the fuel. Others may fall, by gravity, to the bottom of the pool or the low points of the primary coolant system or be trapped in the special devices.

The most common locations are as follows:

- Thermal sleeves of the pressuriser.
- Steam generator packing glands.
- Valves of the primary cooling system.
- etc.

After connection of the Residual Heat Removal System (RHRS), some hot spots may migrate into this circuit and be deposited or fixed. The most common locations are: the pumps, heat exchangers and valves of the circuit.

An underwater pool cleaner should pass through the pool out after discharging. In this case, particularly high equivalent dose rates, equal to or greater than 1 Sv per hour, measured in contact with the filters, represent the last indicator of the possible presence of hot spots, before draining of the pools.

Since no warning signs have been identified yet, to indicate the occurrence of hot spots, it was decided to concentrate on preventive filtering, trapping hot spots as close as possible to their source to eliminate them.

3.1.4 *Preventive strategy*

Stellite limitation

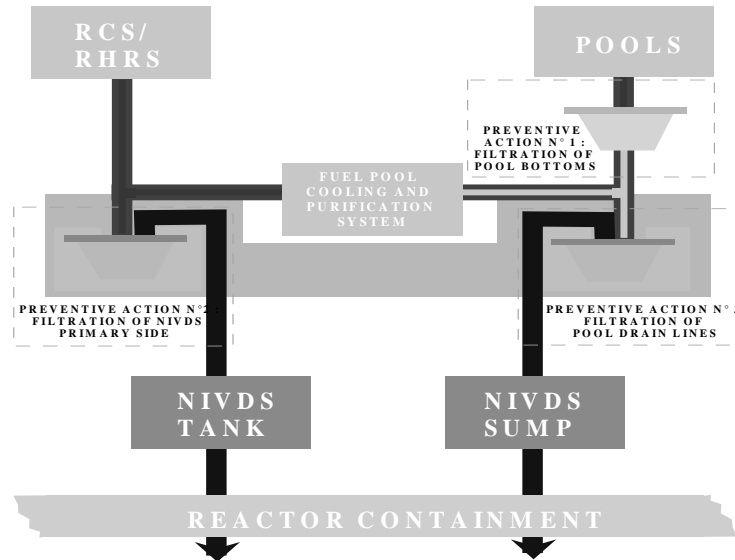
It concerns all the units by performing the following actions:

- limiting the use of cobalt based components in contact with the primary coolant;
- removing stellite particles produced during some maintenance operations (e.g. lapping).

Preventive filtration with specific devices

The simplified diagram below illustrates the principles of the preventive filtration methods which are proposed (Figure 1). It consists on the filtration of all the effluents which could transport hot spots outside the Reactor Building.

Figure 1. Preventive filtration methods



The drains of pools are important routes for hot spots, before they spread through the systems. The installation of fine filters, an initial containment barrier, is proposed for the drain orifices of each pool:

- refuelling cavity located in the reactor building;
- spent fuel pit, located in the fuel building.

Appropriate filters are required:

- truncated sieves;
- 50 μm mesh.

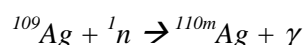
The drain lines of the primary cooling circuit represent the second main vector for the movement of hot spots. As a preventive measure, the installation of sleeves on the main migration channels enables sensitive systems to be filtered. This filtration prevents the spread of hot spots and ensures that they are eliminated.

The filtration system is installed in the reactor building or the fuel building, depending on the shutdown phases, thus enabling all the pools to be treated with the same device.

3.2 *Metastable silver 110 contamination*

3.2.1 *Silver sources*

In some PWR units, the primary and auxiliary systems are subject to metastable silver 110 (^{110m}Ag) contamination, due to silver 109 neutronic activation:



The most likely root causes for silver leading to the observed ^{110m}Ag pollutions may be as follows:

- the neutron-absorbing alloy Ag-In-Cd contained in control rods;
- some silver coated seals.

For French NPP, the major source for ^{110m}Ag is natural silver. Moreover, for the French plants polluted with ^{110m}Ag , the quantity of metallic silver released in the primary circuit could be roughly estimated 1-10 grams.

3.2.2 *Impact on dosimetry*

^{110m}Ag systems contamination can be very penalising since it impacts:

- dosimetry integrated during maintenance operations;
- shutdown schedule;
- control rods management (examination and rejection).

It can represent 5 to 15% of a shutdown total dosimetry.

Auxiliary systems most sensitive to silver contamination are:

- The Residual Heat Removal System (RHRS) heat exchangers.
- The Chemical and Volume Control System (CVCS) non regenerative heat exchangers.
- The purification system downstream the CVCS demineralisers.

^{110m}Ag can contribute to more than 90% of the dose rates around some parts of auxiliary systems.

3.2.3 *Silver behaviour – indicators*

The volumic activity of ^{110m}Ag in the primary coolant stays at a low level (1-10MBq/t) due to the low silver solubility in reducing medium. Activity increases higher than 10 MBq/t indicate a poor silver removal in the CVCS or an important silver source.

During the shutdown, in case of silver pollution, the observed levels of activity increase dramatically (several decade) during and after the oxygenation. This could be explained by:

- Changing from reducing to oxidising chemical conditions.
- Simultaneously by temperature and pH changes.

At the opposite from other corrosion products for which dissolution is maximum at the “oxygenation peak”, silver dissolution shows a trend to go on after the effective oxygenation (dissolved oxygen concentration close to 1 mg/kg).

Generally, the silver peak activity appears 1 to 12 hours later after the Co 58 peak. The Table 2 gives further data about silver peaks for the French plants in the year 2000 during shutdown.

Table 2. ^{110m}Ag , 900 and 1 300 MWe French standardised plant series in 2000

Standardised plant series	^{110m}Ag average	^{110m}Ag maximum	^{110m}Ag minimum
900 MWe	1.47 GBq/t	6.17 GBq/t	0.07 GBq/t
1300 MWe	0.16 GBq/t	0.48 GBq/t	0.04 GBq/t

There is no clear correlation between the ^{110m}Ag activity peak and the other corrosion products. A first level of assessment can lead to consider that RCS is polluted when ^{110m}Ag peak during shutdown is higher than 0.5GBq/t on primary coolant sampled on the Nuclear Sampling System (NSS).

The observation of different and unpredictable behaviour of silver, mostly measured by gamma-spectroscopy and particularly during different cold shutdowns, even in a same plant, can be surprising when the operation schedule seems stable. Measurements demonstrated that it is possible to decrease a lot the steam generator channel head silver contamination and to increase the CVCS exchanger.

Depending on chemical environment and physical properties (pH, redox potential ...) silver would be:

- In ionic form in solution, Ag (I) cation.
- In metallic form, Ag (0), likely under colloidal form (200-600 Å).

Studies, conducted by EDF, demonstrated the essential simultaneous impact of redox potential, pH, and temperature on silver behaviour. In a large range of pH (corresponding to the nominal operating conditions or to the shutdown ones at 300°C, 80°C and 30°C) and in different locations of RCS, CVCS and RHR, Ag(0) and Ag⁺ can be simultaneously found on thermodynamic stability diagram and are very sensitive to the redox variation and the concentration. Actually, it seems that the physical and chemical conditions of the primary coolant set silver mostly on the limits of different existence areas on considering the corresponding pH-potential diagrams.

On the opposite, pH is determinant on colloids behaviour. Repulsion between them is the lowest at iso-electric point, meaning at pH for which the zeta potential is zero.

In the other hand, kinetic behaviour of silver in primary coolant conditions is not known well enough.

Thus, optimisation of its removal is difficult, the preferential deposition of ^{110m}Ag taking place in “cold points” of auxiliary systems exchangers, where thermal gradient is important. This also may elucidate the apparent unoperating conditions of the purification. The goal will become to manage the silver so as to transport it in the form able to be removed by the purification system.

3.2.4 *Preventive strategy*

Control rods non destructive tests

Eddy current and ultrasonic non destructive control rod tests have been performed. Doubtful rods or rods with perforation indications are systematically replaced with either chromium or nitride coated rods.

Purification during end of cycle shutdown

^{110m}Ag purification is performed on a dedicated ion exchanger operating at the maximum possible flowrate [3].

During shutdown, when the filters are replaced, purification must not be suspended. Filters consistent with colloids removal are required (lower mesh with zeta polarised filtration medium). Thus, silver removal is improved and downstream resins pollution may be avoided (poisoning by colloids). If the upstream filter is not redundant, the filter replacement must lead to reduce by half the CVCS flowrate purification to minimise resins pollutions.

The background and the resin features show that macroporous mixed-bed would be the most adapted for soluble silver and silver colloids. The pressure drop must be specially monitored.

Until now, the feedback demonstrated that withdrawing lithium of the primary coolant with a non lithiated mixed bed ion exchanger during shutdown, giving the lower pH, improves the silver removal efficiency. Without fuel cladding failure, lithium can be eliminated starting from the rods drop.

3.3 *Antimony contamination*

3.3.1 *Antimony sources*

In some French PWRs, ^{122}Sb and ^{124}Sb volumic activity peaks have been observed. These peaks were higher than ^{58}Co peaks (table 3). They impacted dosimetry and waste management.

^{122}Sb and ^{124}Sb are two multi gamma emitters with radioactive half-life of 2.7 days and 60.2 days respectively. Thanks to its short radioactive half-life, ^{122}Sb has a low impact on health physics, but it allows to determine antimony sources.

Table 3. Mean and maximum volumic activities encountered at the oxygenation peak

Standardised plant series	Values	^{122}Sb (GBq/t)	^{124}Sb (GBq/t)
900 Mwe first series	Mean	21	14
	Maximum	200	110
900 MWe	Mean	25	18
	Maximum	160	140
1 300 MWe	Mean	14	11
	Maximum	70	55

Two major sources have been identified in French PWRs:

- 1) Pumps bearing wear in the Boron Recycle System. These bearings are made with graphite impregnated with about 10% of antimony.
- 2) Beryllium-Antimony source rod failure, but there are source rods only for the first reactor cycles.

A few activated antimony grams can explain such peaks of ^{122}Sb and ^{124}Sb . So, calculations demonstrate that about 7 grams can lead to a ^{124}Sb peak of 100 GBq/t.

3.3.2 Impact on dosimetry

The important antimony released activity has an impact both on dosimetry and waste management.

^{124}Sb deposited activities on the out-of-core surfaces increase a lot as it is shown in Table 4.

Table 4. RCS recontamination after a ^{124}Sb peak of 70 GBq/t

^{124}Sb (GBq/m ²)	Before H ₂ O ₂	After H ₂ O ₂ injection
Hot leg	0.29	1.02
SG tubing	0.32	0.95

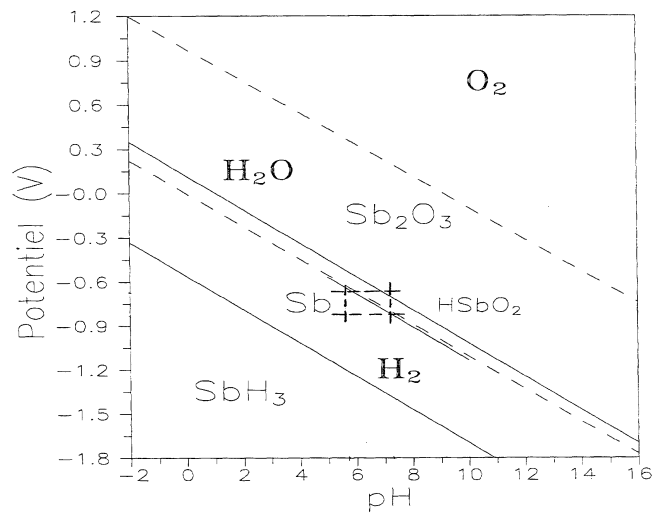
Thus, ^{124}Sb contribution to the dose rates can reach 10% in some parts of circuits. And the dosimetry for maintenance during a plant shutdown can increase to about 5% because of the ^{124}Sb contamination.

3.3.3 Antimony behaviour

EDF studies, elaborating pH-Potential diagrams at normal operating conditions (300°C) and at forced oxygenation conditions at 80°C confirmed the behaviour observed on plants.

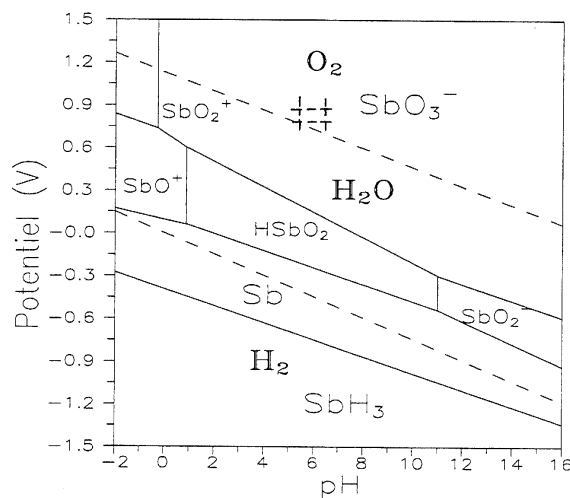
It seems that, in nominal conditions at 300°C, metallic antimony is likely the more stable in aqueous reducing medium (Figure 3).

Figure 3. **Simplified Pourbaix diagram for antimony at 300°C, the dotted line limits the RCS operating conditions**



For shutdown conditions at forced oxygenation, it seems that the more stable species is at the limit of the SbO_3^- area in aqueous solution. Eliminating antimony as well as possible becomes easier in this zone (Figure 4).

Figure 4. **Simplified Pourbaix diagram for antimony at 80°C, the dotted line limits the RCS shutdown conditions at forced oxygenation**



3.3.4 Preventive strategy

Pumps bearing replacement

To avoid antimony pollution, the incriminated pump bearings have been replaced by antimony free pump bearings. As indicated in the table below (Table 5), this allows to make the ^{124}Sb peaks decrease.

Table 5. Bearing replacement consequence during the 14th cycle

Cycles	13	14*	15
^{124}Sb (GBq/t)	100	60	20

Shutdown purification

During oxygenation, antimony seems to be on a soluble form ($< 0.45 \mu\text{m}$). When an antimony pollution occurs, to improve purification by the ion exchange resins, reactor coolant lithium content has to be lower than 1 ppm [4]. In order to obtain acidic conditions, purification can be carried out thanks to non-saturated lithium cation bed demineralizers of the Boron Recycle System. Nevertheless, it should be noted that an acidic reactor coolant seems to increase antimony deposition on the primary circuit surfaces [4].

As for silver behaviour, to manage an antimony pollution, the operators face some difficulties linked to the different physical and chemical conditions of the RCS and the auxiliary systems. Except for removing the root cause, the objective consists in the optimisation of the form of the antimony to enable the purification line to remove the pollution.

4. Conclusion

EDF plants faces radioactive pollutions (i.e. ^{60}Co , $^{110\text{m}}\text{Ag}$ and antimony) leading to significant increases of shutdown dosimetry in affected units (10-30%) or increasing the contamination risk for personnel. Early detection in operation as well as during shutdown requires a good knowledge of their behaviour to set up appropriate solutions.

Replacing critical materials as soon as it will be possible should limit the source term. Limiting pollution in circuits need to implement chemistry and shutdown procedure to minimise over-contamination.

It is obvious that implementing specific chemistry and optimising purification features (filters, resins and flowrate) warrant limiting the effects of this type of pollutions.

That's why studies are yet in progress to propose operators a chemical policy and adapted procedures for each type of radioactive pollution able to reduce the impact on dosimetry.

On the other hand, a new specific gamma spectrometer – able to characterise on the field the nature of radioactive deposits, in order to improve diagnostic, is being developed.

References

- [1] F. Dacquit *et al.* Corrosion Products transport in French PWR'S during cold Shutdown. International conference "Chemistry in water reactors" SFEN, Avignon, 22-26 April 2002.
- [2] A. Rocher *et al.* Program to eradicate hot spots (^{60}Co) carried out in French PWR Units. JAIF International Conference on Water Chemistry, 1998.
- [3] M. Berger *et al.* Policy and Good Practices in Électricité de France PWRs, for Primary Coolant Purification during Refuelling Cold Shutdown, "Low Level Waste and Liquid Separation Technology in Nuclear Applications". WANO Workshop, Varbergs (S), 2001.
- [4] G. Gaudard *et al.* Improvements of Primary Coolant Shutdown Chemistry and Reactor coolant cleanup system. International conference "Chemistry in water reactors" SFEN, Avignon, 22-26 April 2002.

FUEL DECONTAMINATION AT RINGHALS 1 WITH THE NEW DECONTAMINATION PROCESS ICEDEC™

E. Fredriksson, R. Ivars, A. Rosengren

Westinghouse, Sweden

G. Granath

Ringhals AB, Sweden

Abstract

The new fuel decontamination technique ICEDEC™, which has been developed by Westinghouse, is based on abrasion of fuel crud with ice particles. A mixture of ice and water is led continuously through the fuel assembly, which is placed in a specially designed fuel decontamination container connected to a closed loop recirculation system. The ice particles scrape off the loose crud from the fuel surfaces and a mixture of crud and water from the melted ice is then led to a filter unit where the crud is separated from the water.

In this paper results of fuel decontamination tests of two-year-old and spent fuel assemblies during spring 2001 at Ringhals 1 are presented.

The fuel crud was only released when ice particles passed through the fuel assembly and stopped within ten seconds after the feeding of ice particles had ceased. The activity release from the fuel could thus be performed in a controlled way making the process easy to manage and survey.

Activity measurements confirmed that about 50% of the loose crud was removed from the fuel surfaces of the two-year-old assembly. Fuel inspection after the decontamination process showed no influence on the fuel integrity.

Furthermore, no enhanced personnel radiation dose was involved with the fuel decontamination compared to normal fuel services.

Introduction

One of the major concerns in operating a nuclear power plant is to minimize the radiation exposure to the personnel. The last decades a lot of efforts have been made for that purpose. Among all different kinds of measures to reduce the mandoses in LWRs fuel decontamination is the only one that directly attack the origin of activity buildup in the power plant – the fuel crud.

In the core the fuel crud elements are neutron activated forming radioactive nuclides. Radioactive (and inactive) nuclides – especially those in the outer layer of the crud, the loose crud – are to some extent released to the reactor water and subsequently distributed to different system surfaces of the power plant causing activity buildup. By removing the loose crud from the fuels

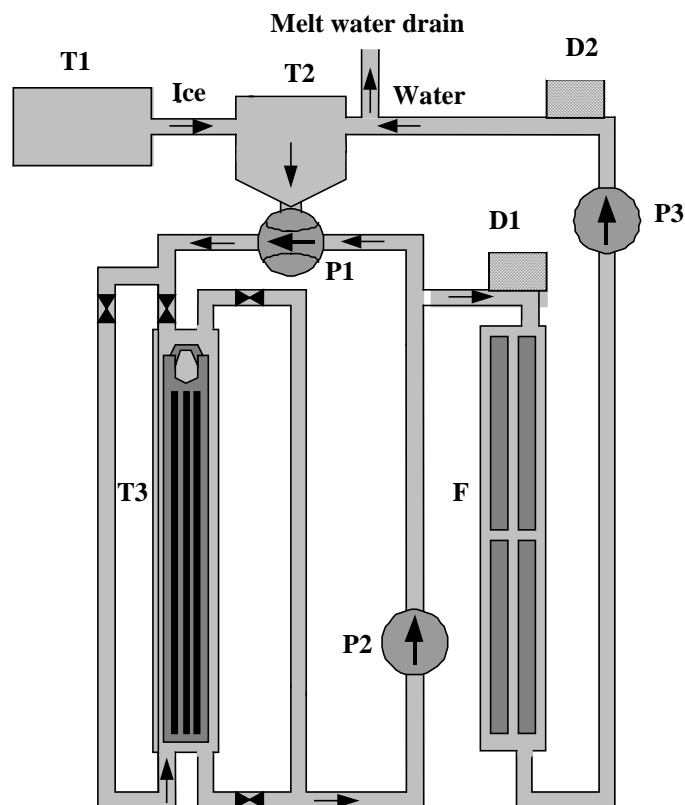
surfaces in a controlled way by a fuel decontamination technique such as ICEDEC™ there will be less nuclides left in the loose crud that are able to release. Hence, there will be less activity buildup in the plant. For that purpose the fuel decontamination method ICEDEC™ was developed by Westinghouse Atom, Sweden.

The development of the ICEDEC™ equipment and decontamination procedure was performed in several steps; feasibility studies, laboratory tests, full scale test of fuel dummies with synthetic crud and recently, decontamination tests of spent and two year old fuel assemblies [1,2]. In this paper results from the full-scale tests of the spent and the two year old assembly are presented.

The ICEDEC™ equipment and procedure

The ICEDEC™ equipment consists of a closed loop recirculating system as sketched in Figure 1. The system contains three units; a decontamination unit for decontamination of the fuel assemblies, a filtering unit which filters the removed crud and a control unit for monitoring and control of the decontamination process.

Figure 1. Sketch of the ICEDEC™ equipment



The decontamination unit contains a conventional ice machine connected to an ice tank (T1), an ice/water mixing chamber (T2), an ice/water pump P1, a decontamination container (T3) for the fuel assembly, a circulation pump (P2) and an auxiliary circulating pump (P3) for recirculating water. In addition, there are valves and a number of temperature, pressure and activity gauges.

The maximum flow of the circulation pump is 20 kg/s, which corresponds to the maximum flow through an assembly in the reactor. The ice particles are made from deionized water and are 3-5 mm in size. In the mixing chamber T2, the ice particles are mixed with filtered water, recirculated via P3. The ice/water slurry thus created is then introduced into the decontamination container, T3, either downwards or upwards through the assembly. Before entering the decontamination container the slurry is passing a net with the hole diameter of 3.7 mm, hence maximizing the ice particles to this size.

Fuel crud is then removed by abrasion when the ice particles are scraping the fuel surfaces. One part of the slurry, now consisting of removed crud and water, is led to the filter unit (about 25% of the flow) and the rest back to the decontamination container.

The filter unit, which is designed to withstand highly radioactive crud, separates the crud from the water. One part of the filtered water is recirculated in the ICEDEC™ system and another part, i.e. the same amount as is introduced to the system via the ice machine, is transported to the pool drain when the activity in the water is below a certain limit.

During the decontamination procedure continuous measurements of total gamma activity is performed by means of BGO scintillation detectors. The detector heads are placed in lead shielded water proof tubes of aluminium, which are mounted in the flow direction at the inlet of the filter unit and at a position after the filter unit, respectively (D1 and D2 in Figure 1). At D1 the water is flowing continuously and at D2 the water is collected in a tank. When the activity at D2 is below a certain limit the residual water is led to the pool drain, otherwise the water is recirculated until the activity has decreased to a value below the limit.

The filter unit contains a stand with room for five filter modules, tubes and valves. The filter modules are dimensioned for more than five kg crud. Four filter modules are arranged in parallel. Each of these contains two sections with the poor size 10 µm and 1 µm, respectively. A fifth module with a poor size of 0,5 µm used for polishing the process water, is connected in series with the others. Each filter module has external dimensions as a BWR fuel assembly. This means that they can be handled with the same equipment and in the same way as the fuel. In addition, it is possible to connect an ion exchange module with ion exchanges resins to the filter unit.

One of several benefits of ICEDEC™ compared to other fuel decontamination methods is that ICEDEC™ is a non-chemical decontamination technique; only deionized water is used. Hence the risk of intergranular stress corrosion cracking or corrosion due to chemical interaction with the cladding and spacer material is reduced.

Decontamination tests at Ringhals 1

Decontamination trials of the spent (five-year-old) fuel were primarily performed as process verification tests and to optimise different parameters (flow, flow direction pressure drop, temperature etc.) of the decontamination procedure. Later on these optimised process parameters should be practiced for decontamination of a two-year-old fuel assembly.

The decontaminations continued for about 10 minutes. Thereafter the ice feeding was stopped and the decontamination container was purged with water during five minutes. When the activity in the system was sufficiently low the equipment was turned off and the fuel was removed from the decontamination container.

When the decontamination tests were completed the ICEDEC™ equipment was system decontaminated with ice and with the decontamination container empty. Before dismounting the equipment activity measurements of different ICEDEC™ components were performed.

During the decontamination procedure on-line measurements of gamma activity from the released crud were performed at positions both before (D1) and after (D2) filtering. Also the temperature and the pressure drop in the decontamination unit was measured. In addition, water samples were taken at positions before and after the filter unit. From process data thus obtained the decontamination fraction DF and the filter efficiency was estimated.

DF was determined from the total gamma activity data at D1 (see Figure 1) as:

$$DF = (A_{\text{before}} - A_{\text{after}}) / A_{\text{before}} \quad (1)$$

Gamma activity measurements of water samples taken at positions before and after the filter unit were used for estimation of the filter efficiency as:

$$\text{Filter efficiency} = (A_{\text{before}} - A_{\text{after}}) / A_{\text{before}} \quad (2)$$

Crud sampling

Crud sampling on rods before and after decontamination was carried out with Westinghouse Atoms equipment consisting of a remotely controlled underwater sampling unit connected to a poolside sample receiving unit and a control unit. The aim of the crud sampling was to determine the amount of loose crud on the fuel and to determine how much of the crud that is removed by the ICEDEC™ procedure during decontamination of the two-year-old fuel.

Both brushing with a nylon brush and scraping with sintered Al₂O₃ was performed. The activity of ⁶⁰Co was then measured.

Since the activity in fuel crud in LWRs is dominated by ⁶⁰Co, this nuclide was used for determining the decontamination fraction according to (1).

Gamma scanning

Gamma scanning of filter units was performed after completed decontamination tests of the five-year-old fuel. The gamma scanning procedure involves continuous movement of the object (usually a fuel assembly) to be studied in front of a collimator slit of a high resolution Ge detector. Processing of the large number of nuclide specific gamma spectra thus generated, results in activity profiles of the object.

In addition, gamma scanning of the decayed five-year-old fuel assemblies was performed both prior to and after the decontamination tests.

On the fuel assemblies the scanning was made specifically of ^{60}Co (since this is the only nuclide from the crud whose activity is large enough to be detected) and of ^{137}Cs .

It must be emphasised that only decayed fuel can be measured since the background radiation from the fuel is too large in newly shut down fuel.

Fuel inspections

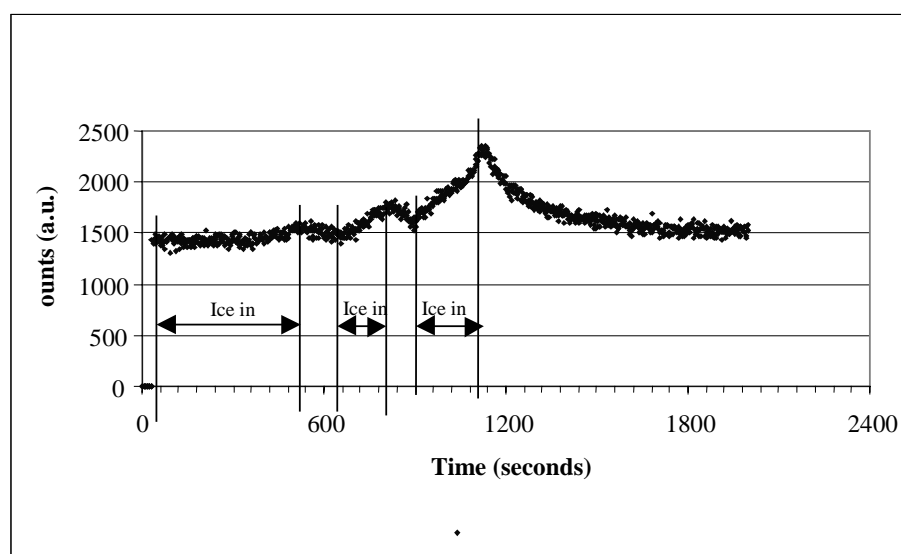
The fuel assemblies were inspected both prior to and after the decontamination procedure¹. The main aim of the inspections was to verify that the ICEDECTM procedure maintains the fuel integrity.

Results

Decontamination tests

In Figure 2 the activity of removed crud (measured at position D1 in Figure 1) is shown as a function of time.

Figure 2. **The gamma activity measured at the inlet of the filter unit during a decontamination test**



From Figure 2 it can be seen that the activity increased when ice particles were admitted into the assembly and decreased when the ice feeding was stopped and only water was passing through the decontamination container. In fact, it was observed that the activity started to decrease within 10 seconds after the ice feeding had stopped.

During the ice feeding the temperature in the decontamination container decreased from 20°C to below 5°C. It was noticed that the activity began to increase when the temperature had decreased to about 6°C.

From the activity data it was concluded that 34% of the total activity was removed from the two-year-old fuel. This corresponds to a decontamination fraction of the loose crud of 53%.

Since the five-year-old fuel assemblies were used mainly as verification of the decontamination procedure there was no relevant data for estimating DF.

When the highest activity was observed at position D1 during the decontamination process the filter efficiency was between 90 and 95%. At lower activity a lower filter efficiency was estimated. This decrease in filter efficiency with decreasing activity is probably due to the presence of particles of sizes less than 0,5 µm and colloids in the removed crud that pass through the filter modules and ion exchange resin.

In addition to the results from the ICEDEC fuel decontaminations mentioned above it was experienced that the equipment was ease to handle; decontamination and dismantling of the ICEDEC equipment after completed fuel decontaminations proceeded without any complications. Furthermore it was concluded that fuel decontamination with ICEDEC does not involve higher mandoses than fuel services during a normal refuelling outage.

Crud sampling

The percentage of loose crud in the two year-old fuel was determined to about 65% and in the five-year-old fuel about 8%.

From scraped crud samples of the two-year-old fuel it was determined that 30% of the total activity was eliminated by the ICEDEC™ procedure. This corresponds to a decontamination fraction of loose crud of 46%. The decontamination fraction of loose crud determined from brushed crud samples of the two-year-old fuel was 42%.

Gamma scanning

Gamma scanning of the filter unit showed that about 85% of the activity in the filter originated from ⁶⁰Co. It was also concluded that more than 80% of the activity from the removed crud was trapped in the 10 µm filter.

Gamma scanning of the decayed (for several months) five-year-old fuel assembly showed that this method can be used for detecting possible displacements of fuel spacers [1]. Owing to the high background radiation from the fuel it is not relevant to use gamma scanning for estimating the decontamination fraction.

Fuel inspection

The conclusion from the tests and inspections of the fuel assemblies was that ICEDEC™ fulfils the main criterion of fuel decontamination, i.e. maintaining the fuel integrity [1]. Only crud was

removed from the fuel surfaces and the oxide layer remained intact. The two year old assembly was then put back into the core for further irradiation.

Discussion and Conclusions

From crud sampling *loose crud* was defined as the amount of crud that is obtained by brushing with a soft nylon brush. The estimation of the decontamination fraction for the two-year-old fuel assembly from the total activity of removed crud and from the ^{60}Co activity of crud samples was then based on that definition. The DF varied between 42 and 53%. It must be emphasised that these results are based on a single assembly, which is too few for drawing any further conclusions.

Questions of great importance that arise when discussing decontamination fractions are: What is the real definition of loose crud if we mean crud that is released from the fuels surfaces into the reactor water and is responsible for the activity buildup in the plant? How much of the crud must be removed by fuel decontamination so that the release of the remaining crud is minimised? There are no answers to these questions yet but experiences from future fuel decontaminations will most likely contribute to the understanding of this subject.

From the ICEDEC fuel decontaminations tests of the spent five-year-old fuel and of the two-year-old fuel it was experienced that:

- ICEDEC fulfils the criterion of maintaining the fuel integrity.
- Only crud is removed by ICEDEC – the oxide remains intact.
- The fuel decontamination procedure is easy to control.
- crud is only removed when ice is introduced to the fuel assembly.
- the removal of crud ceases within 10 seconds after the ice feeding is stopped.
- Fuel decontamination with ICEDEC does not involve higher mandoses than fuel services during a normal refuelling outage.
- The equipment is easy to clean and dismount after completed fuel decontamination.
- The filter efficiency was between 90 and 95% at the highest activity.

References

- [1] L von der Burg, E Fredriksson, R Ivars, A Rosengren, “Fuel decontamination – One way to drastically lower mandoses in LWR”, Trans. ENC’98, World Nuclear Congress, October 25-28, 1998, Nice, France.
- [2] L von der Burg, E Fredriksson, R Ivars, A Rosengren, “, “Fuel decontamination – One way to drastically lower mandoses in LWR”, Proc. Jahrestagung Kerntechnik, Annual Meeting on Nuclear Technology, May 1999, Germany.
- [3] E Fredriksson, R Ivars, A Rosengren and G Granath “Inspection of fuel assemblies decontaminated with the ICEDEC™ technique”, To be published in Jahrestagung Kerntechnik, Annual Meeting on Nuclear Technology, May 2002, Germany.

STEAM GENERATOR REPLACEMENT FROM ALARA ASPECTS

I. Terry

Framatome ANP, Germany

B. Breznik

Krško NPP, Slovenia

Introduction

This paper is going to consider radiological related parameters important for steam generator replacement (SGR) implementation. These parameters are identified as ALARA related parameters, owner-contractor relationship, planning, health physics with logistic services, and time required for the replacement. ALARA related parameters such as source or initial dose rate and plant system configuration define the initial conditions for the planning.

There is room to optimise work planning, managerial procedures and also the staff during the implementation phase. The overview of these general considerations is based on the following background: using internationally available data and the experience of one of the vendors, i.e. Siemens-Framatome, and management experience of SG replacement which took place at Krško NPP in the spring of 2000.

Generally plant decisions on maintenance or repair procedures under radiation conditions take into account ALARA considerations. But in the main it is difficult to adjudge the results of an ALARA study, usually in the form of a collective dose estimate, because a comparison standard is missing. That is, very often the planned work is of a one-off nature so comparisons are not possible or the scopes are not the same. In such a case the collective doses for other types of work are looked at and a qualitative evaluation is made.

In the case of steam generator replacement this is not the case. Over years of steam generator replacements world-wide a standard has been developed gradually.

The first part of the following displays an overview of SGR and sets the Krško SGR in perspective by applying dose analysis. The second part concentrates on the Krško SGR itself and its ALARA aspects.

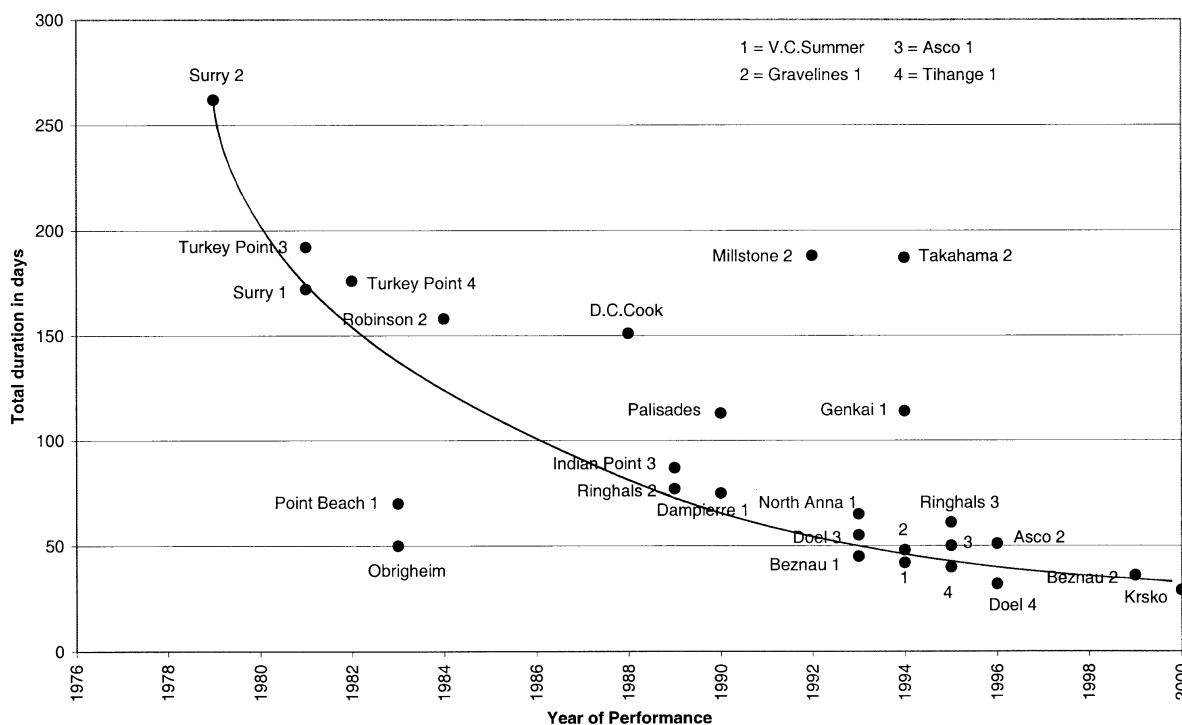
Replacement times and doses

During the last ten years there have been more than thirty steam generator replacements performed at PWRs in the world. From the data of ISOE the collective dose per steam generator was in most cases from 0.3 to 0.8 man-Sv. Collective dose is a rough indicator of the project planning and of overall occupational exposure control.

The procedures have become very similar and collective doses and replacement times have tended to reach a plateau around which the replacements fluctuate. Figure 1 illustrates how the replacement times have dropped drastically from the first beginnings down to a near-constant level. The general curve drops to a plateau around 40 days. We will return to this subject later in connection with the theme of ALARA.

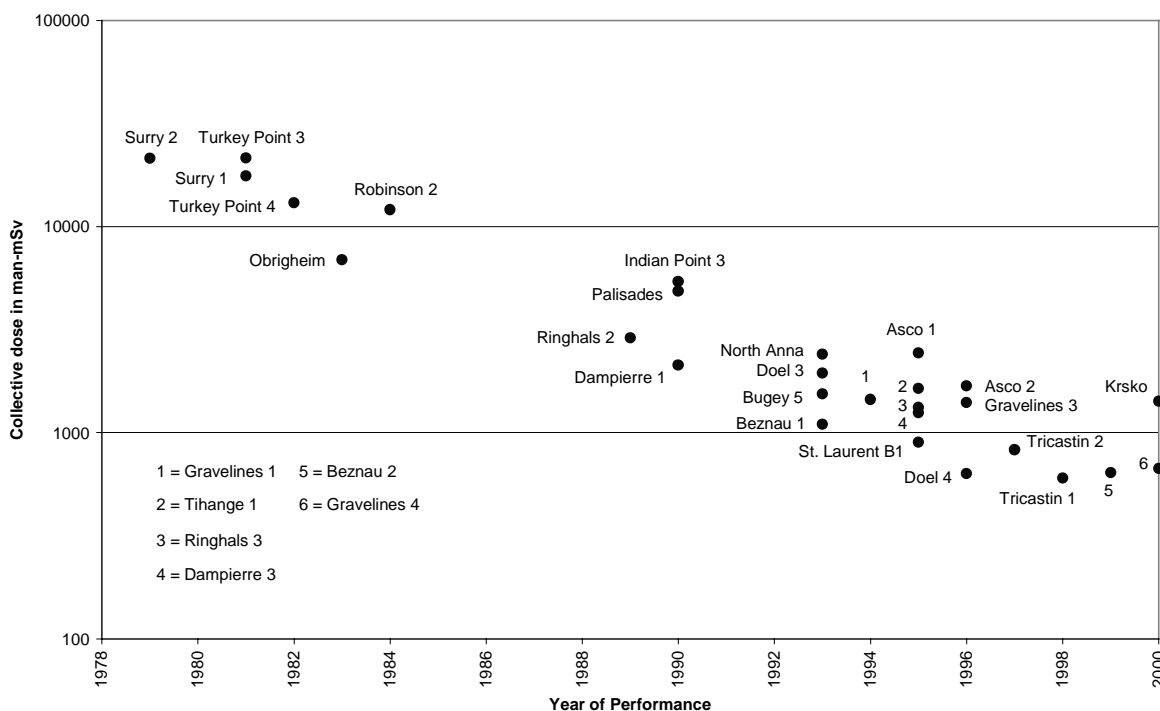
Analysis of the data for such steam generator replacements provides a unique opportunity of making a true ALARA judgement. The work scopes have become very much the same and therefore the number of staff required and the number of hours required are very comparable.

Figure 1. Reduction of SGR times



Let us now look at Figure 2 for the case of collective dose in replacements as conducted in the USA and Europe. As can be seen the doses in the last twenty years have dropped, with one or two exceptions, by a great deal. A plateau for the dose can be perceived by eye to lie around 1200 man-mSv.

Figure 2. Collective dose trend with time



Assessment of management procedures

The data, as displayed in Figure 2, are used to make various comparisons on the world market where competition is rife. But is it adequate? It is certainly not a correct comparison from an ALARA stand-point. Why not, since the scopes are assessed to be the same?

There are three important factors which control the collective dose that have little to do with direct ALARA management procedures. These are

- the primary circuit dose rate;
- the number of steam generators;
- the amount of shielding.

The last item can be misunderstood. This is indeed a very important ALARA measure. But ALARA management procedures are required only to install and to remove it. In other words a great deal of lead reduces the need for management. Once it is installed its very material mass influences the collective dose but this has nothing to do with management procedures.

In order to make an ALARA assessment of managerial procedures these three factors have to be filtered out of the collective dose to enable proper ALARA management comparison.

We have decided to define a quantity called Figure of Merit or FOM for short, borrowed from a method related to Monte-Carlo calculations. Turning to our case the FOM also becomes an artificial factor which includes the three previously named factors independent of ALARA managerial

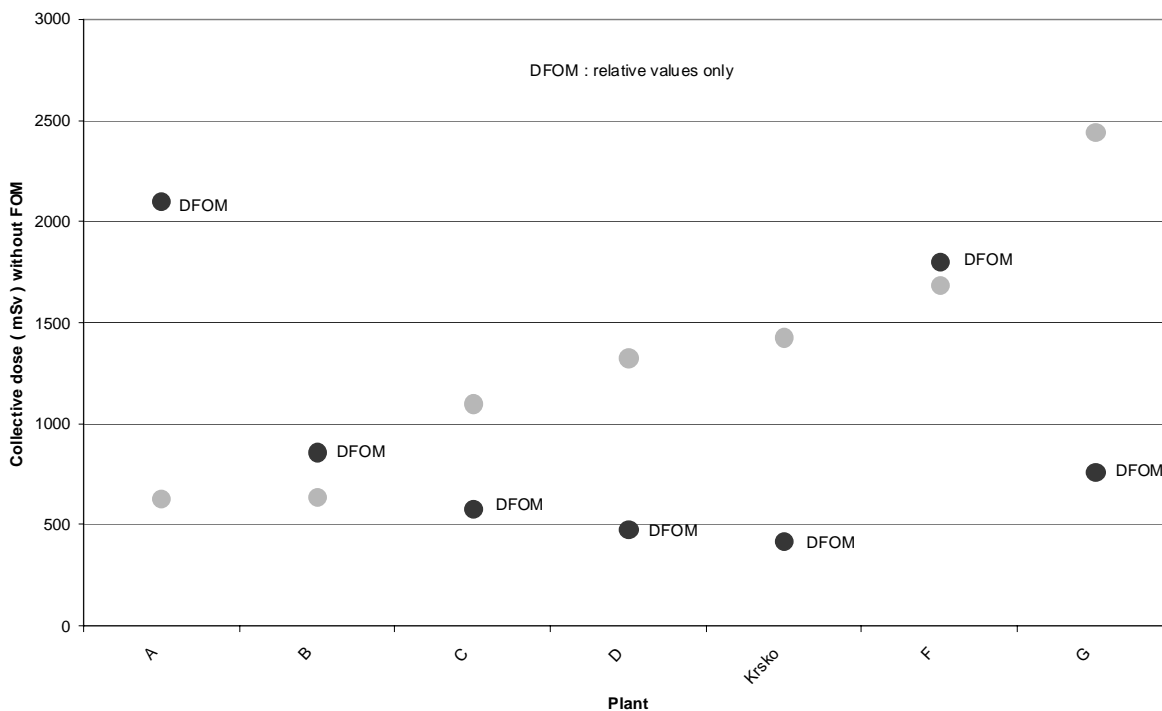
work procedures. Thus we have defined a FOM for filtering the published collective dose that we have seen in Figure 2. The resultant dose is then weighted with the FOM to produce a new weighted dose, a dose FOM or D_{FOM} , which reflects more the managerial ALARA aspects rather than physical conditions over which no control has been exerted. To make this more clear:

$$D_{FOM} = \frac{\text{Total collective dose} \times \text{Amount of lead}}{\text{Number of SG} \times \text{Primary dose rate}} = \text{Total collective dose} \times \text{FOM}$$

For example, if the primary circuit dose rate is small and massive shielding is installed then the FOM will become very large and the D_{FOM} also becomes larger. This presents then a slightly different picture of the collective doses by producing a proper comparison of efforts made from the managerial side. If the D_{FOM} is small than this reflects good management practices.

Figure 3 takes some of the more recent SGR operations shown in Figure 2. The SGR operations have been taken for which the authors have been directly or indirectly involved and therefore for which exact data are known. The collective doses have been modified by the non-physical FOM factor. The number is an index of the application of the ALARA principle by managerial processes. This is a first-time comparison showing true managerial skill in controlling dose in the spirit of ALARA. As can be seen the position of Krško has changed in the assessment. Indeed in this comparison Krško has risen to the top of the ALARA management table. For the higher D_{FOM} plants conditions were such that not so much management was required.

Figure 3. Collective dose analysis illustrating ALARA management



Let us now turn our attention to the Krško Steam Generator Replacement itself and examine some of the management aspects which produced this successful replacement.

Perhaps we should first see what SGR involves with regard to radiation exposure. The main activities with regard to radiation are:

- construction of scaffolding next to the primary loops;
- removal of thermal insulation from primary loops and SG;
- installation of shielding;
- clamping of primary piping to hold it in place;
- cutting of primary piping;
- decontamination of remaining pipe ends;
- machining the new weld lips on the pipe ends;
- welding of new SG to pipe ends;
- general pipe works for auxiliary systems in the loop rooms;
- cleaning activities;
- health physics.

All in all over a thousand single work activities had to be co-ordinated and the doses of more than 400 persons controlled daily. This was a difficult task which required close collaboration.

Owner-contractor collaboration

A good example of owner-contractor collaboration can be presented in the case of the Krško steam generator replacement project (SGRP). Owner-contractor relationship was based on the technical specifications of SG replacement prepared by Krško plant. These had included requirements for the total collective dose of the replacement from one to two man·Sv for two loop plants and preparation of the radiation protection plan, ALARA plan, shielding plan, lay-down area and waste management plan. The contractor was requested to provide radiation protection liaisons in each shift to be an interface between the contractor's technical staff and plant radiation protection organisation. Each organisation had its own SGRP Supervisor who had been responsible also for preparation work.

The plans mentioned above were prepared by the Consortium Siemens-Framatome, and then reviewed and approved by the Krško plant. The ALARA plan included brief descriptions of each activities with the appropriate flow diagrams, if necessary. All radiation protection related plans were then presented to the plant ALARA Committee, for feedback if necessary.

For specific tasks, such as for measurements, scaffolding, insulation removal, clamping, cutting, pipe decontamination, cleaning and shielding, detailed RP instructions were prepared by the planning team, also composed of task managers, and included in the ALARA work planning forms following plant procedures.

SGR ALARA review

Unlike other SGR projects, Krško SGRP had an extremely short replacement time. This was 29 days to Operational Delivery. In other earlier SGR projects, depending on scope, the replacement times lay more by 40 days, as mentioned at the beginning. This short SGR time, which although contractually arranged, had negative effects on ALARA towards the project end. This indicates, too, that further efforts to reduce replacement times could lead to higher doses. An example of this is if more activities have to be performed in parallel towards the end of the SGR. At this stage shielding is being removed prior to hand-over. Thus more dose is generated than would be case with shielding still in place.

One boundary condition for the Krško SGR was the primary circuit dose rate, which was very high. In the unshielded state the contact dose rate was 3.4 mSv/h and 8.6 mSv/h when drained. In view of this fact the collective dose was held to an acceptable level which was only approx. 30% above another SGR (also two loop) where the dose rates were half these values. But we have already seen this comparison.

The collective dose estimate itself was obtained using the Dosiana software which requires large amounts of input for manpower and dose rates. These data are weighted by occupancy factors, which are based on experience, and decide the actual time spent in radiation fields. A repeat run on-site, using the actual dose rates measured after shutdown, showed no major deviation from the prediction, as based on previous years measurements.

SGR doses were analysed according to job codes used during performing the jobs. Because exact planning of work is not always possible and also corrective actions can not be predicted, then an estimate accuracy in a band of around $\pm 10\%$ can be regarded as exact and around $\pm 20\%$ as acceptable.

Some examples of good ALARA practice

Experienced personnel were engaged for the shielding. Training for auxiliary personnel on inactive piping in the water processing plant took place in order to ensure that installation and securing of shielding was almost routine.

Scaffolding work was speedily carried out to avoid dose. This resulted in smaller man-power needs, by a factor of at least three, than that expected. It shows the advantage of using a trained team with good planning for the required scaffolds in advance.

To prevent cross contamination of large items special clean areas were introduced in which no shoe covers were required. As radioactive items were removed this good practice was extended by issuing instructions in the form of ALARA Cleaning Regulations. Thus these clean areas were extended to the operation deck in the containment and then to the Auxiliary Building.

Dose-following was on a daily basis in order to compare actual doses with the predictions. This enabled timely intervention to modify procedures if it was shown that conditions had changed and would be producing unexpected doses if the work was continued unmodified.

Health Physics Management – Krško SGRP

Health physics team of Krško plant conducted operational radiation protection. In each shift there were five health physics (HP) technicians, in the morning shift in addition two technicians and two engineers most of the time. On comparison to other similar replacement projects the number of technicians and engineers was optimised to a very minimum. HP shift leaders and engineers attended training sessions for a few SGR activities. The plant contracted some HP technicians from abroad already experienced in SGR, others were taken from the local Institute for Occupational Health.

The responsibilities and cooperation between NEK and the Consortium were defined in a Project Radiation Protection Plan, as mentioned before. Health physics activities were planned daily by the Radiation Protection Lead Engineer (who acted also as NEK's SGR Supervisor) and by the Consortium Radiation Protection Supervisor. Both supervisors participated in the daily project management meetings.

The use of experienced health physics liasons from the local Institute made a positive contribution to ALARA. These people had been trained for SGR by the Consortium and were fully familiar with the Krško plant and procedures. They ensured that the needs of the teams (protective clothing, measurement services) were fulfilled well in advance.

Conclusion

In conclusion, this paper has provided a method of evaluating ALARA management in SGR operations and named those managerial aspects which lead to keeping occupational exposure low.

APPLYING ALARA FOR REPLACEMENTS OF STEAM GENERATOR FEEDWATER DISTRIBUTION PIPES AT PAKS NPP

G. Volent, I. Metzger
Paks NPP

1. Introduction

The Paks Nuclear Power Plant Ltd. operates four VVER-440 type reactors. Unit 1 has been operating since 1982 and the fourth Unit connected to the grid in 1987. The role of Paks Nuclear Power Plant is decisive in the domestic electric power generation. The share of the nuclear energy in the electric power production of Hungary during the period from 1988, once the Unit 4 was put into operation, is constantly more than 40%.

Similarly to other VVER-440 type NPPs erosion corrosion damages of feedwater distribution pipes were observed at Paks NPP in 2000. An extensive steam generator feedwater distribution pipes replacement programme was started at Paks NPP in 2001. In the frame of this programme the replacement of feedwater distribution pipes was carried out on 14 steam generators in 2001.

The steam generators that are used at VVER-440 type NPPs are horizontal shapes. Each VVER-440 unit has six steam generators. The feedwater distribution pipes are inside the steam generator. The erosion corrosion damages are observed especially on so-called T-connection. The dose rates inside the steam generators, that we measured during outages in 2000, were 5-8 mSv/h on Unit 1, 9-16 mSv/h on Unit 2 and 8-20 mSv/h on Unit 3. A significant part of the replacement work can be only performed inside the steam generator. Based on our own experience the workers all together should spend 50 hours inside steam generator during the replacement.

In the view of the extent of the work and the dose rates at places where work was to be carried out, it was clear from the start that a relatively high collective dose had to be taken into account. The ALARA approach played important role in preparation phase as well as in implementation phase.

Applying ALARA principle means that a work management approach must be adopted that considers all the factors contributing to radiation dose, identifies and co-ordinates the actions which can be implemented and analyses their dose reduction effectiveness in relation to their respective cost. At Paks NPP we followed this approach as much as possible during the replacement of feed water distribution pipes.

2. Preparation of replacement of feedwater distribution pipes

2.1 *First phase*

The preparation of replacement work has been started approximately 8 months before when the first replacement was performed in 2001.

In the frame of first phase of preparation we collected and analysed our own and some other VVER-440 type NPPs' experience in connection with replacement of feedwater distribution pipes. We considered the procedure of replacement and all aspects of work (protective options, schedule of work, working environment, tools and training) in order to optimise the duration of exposure, the number of people exposed as well as the dose rates. In this phase of preparation decision aiding technique was also used. As per the authority approved Plant Radiation Protection Code of Paks NPP differential cost-benefit analyses was used and a monetary value of 25000 HUF (~ 100 USD) per man mSv saved was applied.

At the end of the first phase of preparation the radiation protection experts in co-operation with experts of maintenance, technical support and chemistry completed an ALARA report. This report among others included the results of cost-benefit analyses and the preliminary dose plan of the replacements. Based on the cost-benefit analyses the report suggested the following measures to optimise the dose rates inside the steam generators:

1. one or two cycle decontamination (the number of cycle is depending on the dose rate in steam generator);
2. use of self-shielding effect of water which is in primary and secondary side of steam generator;
3. shielding of hot and cold leg collectors and top layer of heat exchanger tubes inside the steam generator.

The dose rate reduction factors of the above measures were mainly determined by use of earlier measured data. In some cases the factor was assumed by computer code. In the cost-benefit analyses the dose rate reduction factor of one and two cycle decontamination was 2.5 and 6-8 respectively, which figures are based on measured data. The dose rate inside the steam generator will reduce by factor 1.3 if the steam generator secondary side is filled with water to the top of heat exchanger tubes. Filling of primary side of steam generator with water results in 6% dose rate reduction. The suggested 5 tonnes shielding can reduce the dose rate by factor 2, which was calculated by computer code.

2.2 *Second phase*

In the second phase a multidisciplinary team continued the preparation and co-ordinated the preparation activity between all workgroups involved in replacement of feedwater distribution pipes by organisation of regular meetings. During the second phase all aspects of work considered in order to optimise the radiation exposure.

Scheduling

The proper work scheduling is important in maintaining doses ALARA. In case of replacement of feedwater distribution pipes the proper work scheduling and the harmonisation of dose rate reduction measures (water in the primary and secondary side of steam generator) with the conditions of other outage works was very important.

As a result of compromise between the dose rate reduction and prolongation of outage there was no water in primary side of some steam generator during the replacement work that resulted in higher dose rate by 6% than it was planned.

Manpower

The qualified and specially trained manpower availability was also important for that work because of the relatively high number of steam generators where the replacement was planned in 2001. In Paks NPP the plant level monthly dose limit is 6 mSv and the plant level yearly dose limit is 20 mSv. The goal was that none of the workers reach the monthly or yearly plant level dose limits. When we determined the number of workers we took into account this goal.

Working environment

The working condition factors were taken into account during the preparation. The proper light, the convenient temperature and ventilation of secondary side of steam generator were prepared.

Another important subject of preparation that is linked to working environment was to optimise the workload in relatively high dose rate area. We minimised the work inside the steam generator, as much as possible we transferred the steps of replacement work to a low dose rate area.

Training of workers

The Paks NPP has a Maintenance Training Centre at site, which is among others equipped with full size steam generator. The workers received training on that steam generator where the working environment, except the ionisation radiation, is totally same than in the field.

Completing ALARA reports separately for each Unit where the replacement of feedwater distribution pipes was planned during the outage closed the second phase of preparation. These ALARA reports among others included the results of repeated cost-benefit analyses and the detailed dose plan for the replacement works. The dose rate reduction measures practically were not changed. The reports suggested decontamination, use of self-shielding and shielding for decreasing the dose rate inside the steam generators.

3. Work implementation

The replacement of feedwater distribution pipes started on Unit 2 where the feedwater distribution pipes were replaced in 5 steam generators. The implementation of earlier mentioned dose rate reduction measures was usually enough effective, however we had to perform three cycles of decontamination on two steam generators.

After the dose reduction measures the dose rate level inside the steam generators varied between 0.4 and 0.5 mSv/h and it was not higher than 1-2 mSv/h when the workers partly pulled off the shielding.

The multidisciplinary team continued its work and co-ordinated the activity between all workgroups involved in replacement work. The radiation protection personnel provided assistance and advice to workers. The individual and collective dose of workers was controlled continuously by electronic dosimeters. The dose rate were followed and analyzed by supervisors as well as radiation protection personnel.

The collection of feedback data was performed during implementation phase. This collection had two interests:

1. to provide timely feed back and allowing to implement rapid corrective actions;
2. to support the preparation of replacement feedwater distribution pipes on the next Units.

As a result of collection and analyses of feedback data we modified the shaping of shielding inside the steam generator and put slip-way to help access to steam generator and exit. The modifications and the more skills of workers resulted in lower working time in the steam generator and lower individual and collective dose on Unit 1 and Unit 3.

Table 1. Dose results of replacement

Unit	Number of SGs (where the replacement was performed)	Planned collective dose per SG (man mSv)	Actual collective dose per SG (man mSv)
Unit 2	5	40.7	41.5
Unit 1	3	53.4	28.8
Unit 3	6	59.8	23.2

4. Conclusion

The work management approach which was used during the replacement of feedwater distribution pipes helped to consider all the factors contributing to radiation dose and to identify and co-ordinate the actions which can be implemented to optimise the radiation exposure.

REPLACEMENT OF NEUTRON ABSORBERS IN THE SPENT FUEL POOL STORAGE FACILITY OF THE TIHANGE 3: AN ALARA APPROACH

L. Vermeulen
Tihange NPP, Belgium

1. Boraflex in storage racks for fuel elements

Boraflex is a neutron absorbing material used in spent fuel storage facilities to ensure control of criticality, and as such to make it possible to store a larger quantity of fuel assemblies per square metre. It is used in the following nuclear plants in Belgium: Doel 1-2 (some racks), Doel 3 (racks made of 50% boron steel, 50% Boraflex), Doel 4 (100% Boraflex), Tihange 1 (stainless steel racks), Tihange 2 (boron steel racks) Tihange 3 (100% Boraflex racks).

The fuel building at Tihange 3 houses three pools for fuel elements (Annex 1 Figures 1-3): pool D412 containing 1 rack of 3x4 cells (rack T), pool D421 containing 6 racks of 7x6 and 7x7 (racks M, N, P, Q, R and S) and pool D422 containing 12 racks of 6x6, 7x6 and 7x7 (racks A, B, C, D, E, F, G, H, I, J, K and L). A fourth pool D513 is a decontamination pool for inspecting and monitoring fuel assemblies, fitted out and used in this project for HP cleaning of racks, chipping of surrounding plates on four sides, and cutting of plates located in the inter-compartmental spaces.

Boraflex is a supple, composite material (similar to rubber) made by dispersing boron carbide B₄C in a silicone elastomer matrix. Composition by weight (in%): boron (31.5%), carbon (19%), oxygen (22%), hydrogen (3%), and silicon (24.5%). The Boraflex strip is 'sandwiched' between the external wall of the cell and a TIG spot-welded 8 mm stainless steel plate. (Annex 1 Figure 4) The material is therefore in permanent contact with the water in the pool containing boric acid.

2. Degradation of the Boraflex

The irradiated fuel assemblies placed in the racks subject the Boraflex to an intense gamma flux, which causes degradation of the Boraflex arising from the radiolysis reaction, and rupture and rearrangement of the chemical bonds by the gamma rays. These products include: gases (mainly CH₄ and H₂) of soluble silicon (H₂SiO₃), insoluble silicon (in amorphous form) SiO₂, organosilic compounds of (insoluble) boron carbide. It is assumed that the end degradation product of Boraflex is amorphous silicon, which generally occurs in solution in colloidal form (that is to say in grains of less than 1 micron), like activated corrosion products (⁵⁸Co, ⁶⁰Co, ⁵¹Cr).

3. Drawbacks of silicon release

The drawbacks of silicon are the following:

- The capacity of silicon to form gels in the presence of other elements (such as Ca, Mg and Al) results in the filters quickly becoming clogged, which reduces their capacity to effectively filter the radioactive corrosion products.
- The higher concentration in colloidal silicon also causes opacity in the spent fuel storage pool, and even in the reactor pool during servicing.
- The complete removal of the Boraflex leads to an increase in the keff, without boron, of approximately 0.25 (from 0.093 to 1.18), with a 50% reduction in storage capacity. 1 000 ppm of boron is enough to re-establish the margin: 0.97 (limit: 0.98).
- In the primary circuit, a higher silicon concentration in the presence of other elements such as Al, Ca or Mg can result in the formation of zeolites (complex molecules of aluminosilicates, calcium and magnesium) that can precipitate on the cladding of the fuel rods and limit heat transfer.

4. Technical solution

Having examined the various alternatives – including replacing the racks with new ones – the decision was taken to remove the Boraflex from the existing racks and reuse them with another neutron absorber, boron steel or Borated Stainless Steel (BSS). Consequently, all four sides of all the cells must be coated with borated steel to a thickness of 2mm containing 5 mg/cm² of B10. This modification would make it possible to load fresh fuel with an enrichment of up to 5%.

Either a pair of cutters and a chisel method is used as appropriate cutting system. There are two types of jacket. The outer jackets are easily accessible and can be removed manually. The inner jackets are less accessible, the available space being an intercellular channel of cross section is 220 mm x 43 mm or 220 x 50 mm (Annex 1 Figure 5) depending on the jackets and of the same length as that of the racks. First the spot welds are cut on the upper and lower sides with a pneumatic chisel and manual chisels. Then two parallel longitudinal cuts are made simultaneously with cutters fixed on a holder. The jacket and the Boraflex are released. This is the longest phase and is carried out semi-automatically and by remote control. Then the plate and the Boraflex are removed from the cellular channel.

5. ALARA plan and the replacement of Boraflex

Throughout the project, the ALARA principle has been applied in developing tools and choosing working methods. Each phase was then re-evaluated in terms of dosimetry cost in order to make improvements that would mean a reduction in the dosimetry of the operation.

Devising methods and tools – some examples:

- The cover of the decontamination unit acts as a screen for the operators nearby by increasing its thickness.

- The jackets are cut using a remote-controlled cutting tool, which means it is not always necessary to have an operator present near to the rack being processed.
- Due to their shape, material and surface condition the equipment and tools can easily be decontaminated.
- BSS is introduced using a table equipped with a crank system allowing the operator to remain at a distance of five metres from the rack being processed.
- Where possible tools are fitted with extensions enabling the operator to keep their distance from the contaminated article being treated.
- Televisual inspection is carried out semi-automatically, reducing the time taken and therefore operator exposure.

Decontamination

A complete decontamination of pool D513 (in which the decontamination unit is located), as well as underwater suction of the horizontal surfaces of pools D421 and D422, was carried out before starting the project. Each of the racks underwent a two-stage decontamination process:

- First, under water US decontamination is carried out prior to removal from the pool; this makes initial decontamination possible without exposing the operators to a level higher than that normally found at pool level.
- The rack is then decontaminated a second time in the decontamination unit using a high pressure jet of water to remove any transferable contamination. The accessible surfaces are decontaminated manually.

The decontamination unit is decontaminated each time a rack is removed. Other small tools such as chisels coming into contact with the rack are sent as necessary to the decontamination room. All objects coming out of the water are subjected to intensive rinsing: racks, lifting equipment, cameras, screws, long tools etc.

Maximising distance between operator and source

For each phase of the process, the distance between operator and source is evaluated. In order to calculate the maximum distances, all possible practical improvements are included in the study.

Screens

The existing elements in the building, notably concrete and the water in the pool, are used as much as possible to screen the sources of radiation. As far as possible, the racks are treated under water or in pool D513, rather than at pool floor level. Due to its shape and location, pool D513 offers additional protection through the thickness of its cover. Metallic structures equipped with lead plates are available should a rack need to be screened in a horizontal position, in the event that there are any hotspots.

Staff training

All employees receive training and carry out practical simulation exercises in small groups, to become acquainted with the procedures, tools, and work processes. Operations that are carried out in positions where the dosage rate is particularly high are simulated in advance during intensive training in small groups. This practical and theoretical training aims to reduce working time and the risk of confusion when carrying out procedures, which could lead to longer periods of exposure to radiation.

Dosimetric estimate and initial hypothesis

The estimated dosimetric cost for all the operations is presented in table form (Annex 2), making it easier to pinpoint the operations subject to increased exposure. The values are given in man-millisievert. The initial hypotheses for estimating the dosimetric cost are based on information gained from reracking in China, the United States, France and Spain, along with information provided by Electrabel/Tractebel. Exposure times have been calculated on the basis of detailed schedules of treatment of the racks, as have the human resources required for the procedure (in terms of number and position). The values recorded during the treatment of the first rack S serve as a base projection for the treatment of other racks.

Dosimetric follow-up

Staff are equipped with an official dosimetry film and an electronic dosimeter giving a direct reading of the accumulated dose with an audible alarm in the event of the maximum dose rate or total dose being exceeded. The dosimetric reading is linked to a daily dosimetric monitoring system consisting of a dosimetric monitoring form (filled in by the team coordinator) corresponding to the various stages of the operation. Monitoring means that any discrepancies between the estimated values can be detected quickly and the appropriate measures taken.

Plan for remedial action

In the event of the estimated doses being significantly exceeded, a reevaluation of the risks is carried out on the basis of intervention times, positions and actual dose rates. Rack S, used for labelling tools and procedures, has provided actual values, making it possible to revise the estimate of the dosimetric cost, not so much on hypotheses linked to similar activities carried out on other sites, but on the basis of actual values recorded at Tihange 3.

6. VISIPLAN software

VISIPLAN software, used for this project, is an ALARA planning tool that makes it possible to predict the dose equivalents to which staff in a nuclear environment will be exposed whilst carrying out specified operations.

The initial stage of using VISIPLAN involves creating a geometric model of the materials able to absorb gamma radiation, form secondary radiation, and of radioactive sources.

The second stage involves analysis, following calculations, of the dose rate fields in grids (XY, YZ and ZX). These fields can be visualised by means of isodoses. Several dose rate fields can be obtained by speculating on the activity of the sources.

The third stage involves the detailed planning of tasks. Trajectories are defined as a sequence of tasks to be performed in a fixed geometry based on position, duration and description. A scenario is a collection of trajectories to which one or more operators is assigned. The calculation, of a trajectory and a scenario provides a collective equivalent dose value.

The fourth stage is that of dosimetric monitoring when work is being carried out on the rack, followed by modification of the model once work is completed.

VISIPLAN is used interactively. Following many simulations, a collective dose of 13 man-mSv pro racks has been estimated.

7. Replacement of Boraflex with BSS

The different activities will be analysed under the ALARA principle (Annex 2).

7.1 Identification of the rack by means of an ITV camera

The high-resolution auto focus camera with a powerful zoom has been mounted on a pole made up of screwable carbon tubes that are of course very lightweight. The direction of the camera is altered by means of a joystick. There is also a printer and a video recorder. Dosimetric cost: 6 man- μ Sv/h.

7.2 Decontamination via ultrasound

The US cleaning system includes a generator and a transducer or resonator. The generator produces an alternating current of frequencies varying between 20 Hz and 40 Hz. The transducer then changes this current into mechanical vibrations. Two resonators function simultaneously. The water containing contaminated particles is sucked up and then filtered before being piped back into the pool. This operation is controlled visually by the camera. Dosimetric cost: 30 man- μ Sv/h.

7.3 Suction

Following US decontamination, the compartments are cleaned using the Balduf system, which is equipped with a booster designed to suck up any particles remaining after US decontamination. Dosimetric cost: 48 man- μ Sv/h.

7.4 Unscrewing

Each rack in the pool is fixed on a frame at the bottom of the pool by eight anchor points each consisting of a screw, grommets and an anchoring pin. First two sets are pulled out so that guiding columns can be introduced, then the remaining six are removed. Dosimetric cost: 20 man- μ Sv/h(rack S) and 40 man- μ Sv (rack P).

7.5 *Transfer of the rack to the decontamination unit*

To raise a rack out of the water in the pool, there are two stages. The first involves moving the rack onto a submerged platform located near to that being removed. The second stage involves pulling out a sling so as to raise the rack out of the water. Once removed from the water, it is then rinsed thoroughly and then let to drain. Next the dose rates in the vicinity of the raised rack are measured, and any hotspots identified. There is an average dose rate for rack S of approximately 300 $\mu\text{Sv/h}$ with hotspots reaching 11 500 $\mu\text{Sv/h}$, sometimes even 20 000 $\mu\text{Sv/h}$, at the bottom of the rack. Rack P approximately 250 $\mu\text{Sv/h}$ with hotspots reaching 1 200 to 1 500 $\mu\text{Sv/h}$, at the bottom of the rack. Dosimetric cost: 303 man- μSv (rack S) and 101 man- μSv (rack P).

7.6 *HP decontamination*

Cover is placed on the decontamination unit, HP decontamination is then carried out to remove transferable contamination and contamination on the cover of the monitoring platform. Internal dose rate after HP decontamination 100 to 300 $\mu\text{Sv/h}$. Dosimetric cost: 138 man- μSv (rack S) and 171 man- μSv (rack P).

7.7 *Decontamination of external surfaces*

This operation is carried out manually with cloths. External dose rate after decontamination: $\pm 50 \mu\text{Sv/h}$ C Dosimetric cost: 140 man- μSv (rack S) and 112 man- μSv (rack P).

7.8 *Removal of surrounding jackets and Boraflex*

Manual chisels are used to break the solder points and the surfaces are finally cleaned by grinding the residual solder points until they are completely smooth. Dosimetric cost: 699 man- μSv (rack S) and 451 man- μSv (rack P).

7.9 *Preparing the rack before cutting*

- The rack is taken and placed on a tipper. The tipper is used for returning racks since access to the inter-compartmental spaces is easier from below. The racks must be returned in order to begin cutting the internal jackets;
- The rack is turned horizontal, the plates and the rack are inspected visually. The spot welds top and bottom are chiselled, and the indentations are increased at the foot of the racks to make room for knives to be inserted into the inter-compartmental space;
- The rack is turned upside down and placed in the decontamination unit;
- Positioning the holder for the cutting tool.

Dosimetric cost (intervention time): 4 857 man- $\mu\text{Sv}/110 \text{ h}$ (rack S) and 3 393 man- $\mu\text{Sv}/47 \text{ h}$ (rack P).

7.10 Cutting with a holder

With the help of cameras, the cutting head is positioned above an inter-cellular space. As the cut is made, the load is monitored. The holder is removed. The cut plates and the Boraflex are removed. Dosimetric cost/intervention time: 3 499 man- μ Sv/195 h (rack S) and 1 916 man- μ Sv/108 h (rack P).

7.11 Uniform cutting

Dosimetric cost: 652 man- μ Sv (rack S) and 577 man- μ Sv (rack P).

7.12 Chemical cleaning

This stage will become obsolete with the use of non-ferrous chisels. Dosimetric cost: 307 man- μ Sv (rack S) and 163 man- μ Sv (rack P).

7.13 HP cleaning

To remove any remaining traces of Boraflex. Dosimetric cost: 290 man- μ Sv (rack S) and 94 man- μ Sv (rack P).

7.14 Visual ITV inspection

This applies to all channels. Dosimetric cost: 356 man- μ Sv (rack S) and 110 man- μ Sv (rack P).

7.15 Insertion of the BSS plates

The rack is taken, placed on the tipper and turned horizontal. The BSS plates are introduced two at a time into the inter-cellular channels, forming a sandwich. A horizontal guide at each end, a medium length retractor, and a locking mechanism at the very bottom are soldered onto one of the plates. As well as the two horizontal guides, retractor and locking mechanism, six malleable locks are soldered onto the other plate. Once the sandwich is positioned in the inter-cellular space, a spacing tool is introduced between the two BSS plates and shapes the malleable locks by sticking the BSS plates onto the inner walls of the compartments. The locking mechanism is soldered onto the inner compartment wall. Dosimetric cost/intervention time: 2 510 man- μ Sv/77 h (rack S) and 1 709 man-48 h (rack P).

7.16 Insertion of dummy into cells

The rack is turned upside down, placed in the decontamination unit and a mannequin is passed into each of the cells. Dosimetric cost: 72 man- μ Sv (rack S) and 79 man- μ Sv (rack P).

7.17 *Screwing the rack to the pool*

The rack is taken and placed on the bottom of the pool. Dosimetric cost: 80 man- μ Sv (rack S) and 21 man- μ Sv (rack P).

8. Dosimetry

Critical comparison of the key dosimetric phases (> 1 000 man- μ Sv)

Operation		Rack S			Rack P		
		Maximum individual dose (μ Sv)	Total dose (man- μ Sv)	Time taken (h)	Maximum individual dose (μ Sv)	Total dose (man- μ Sv)	Time taken (h)
9	Preparing the rack	1 349	4 857	110	940	3 393	47
10	Cutting	1 247	3 499	195	650	1 916	108
15	Inserting BSS	479	2 510	77	398	1 709	48

The doses registered for rack P are lower than those registered for rack S. There are several reasons for this:

- The time taken for the most high-exposure operations is less (due to better knowledge of the operations to be carried out and stricter compliance with instructions), and so in particular is the time taken to prepare the rack, cut and insert the BSS.
- The operators, knowing that the dosage rate was higher for certain operations, subconsciously worked more quickly.

Improvement for the next racks

- When removing the rack out of the water, precise localisation of the hot spots, followed by efficient local decontamination.
- Optimising screens.

References

Projet Boraflex Tihange 3: Dossier ALARA. (TECHNUBEL: X. Bairiot).

VISIPLAN ALARA Planning Tool. (SCK-CEN – Belgium: F. Vermeersch).

Le remplacement du Boraflex dans les râteliers de stockage de Tihange 3. (GRAMME: F. Migeot).

Application et évaluation du logiciel VISIPLAN sur le site nucléaire de Tihange lors du chantier Boraflex. (UCL-Louvain – Belgium: N. Druet).

Annex 1

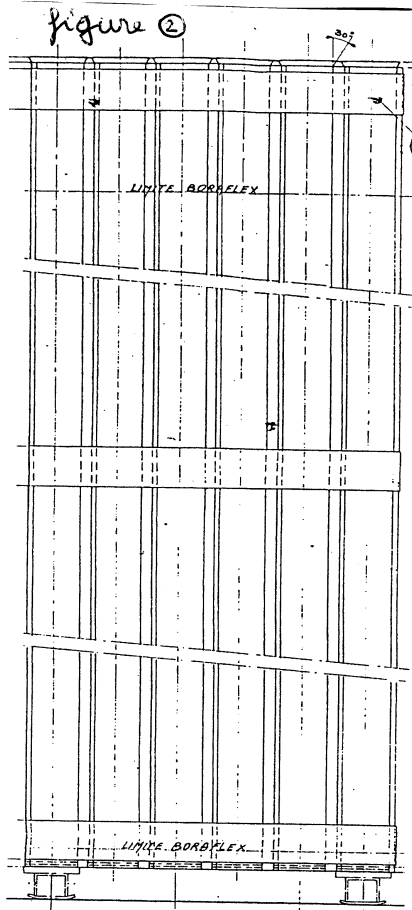
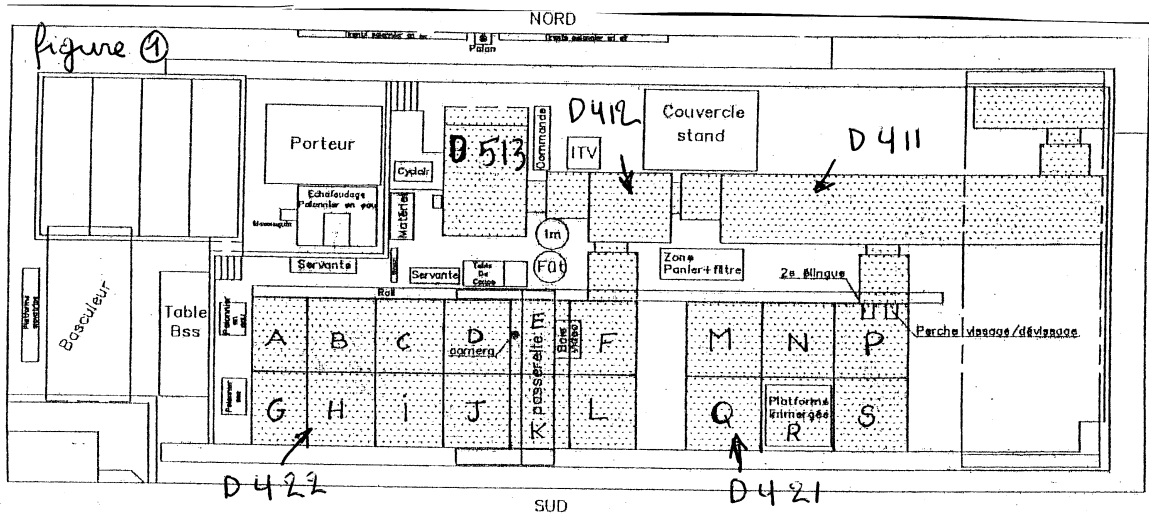


fig ③

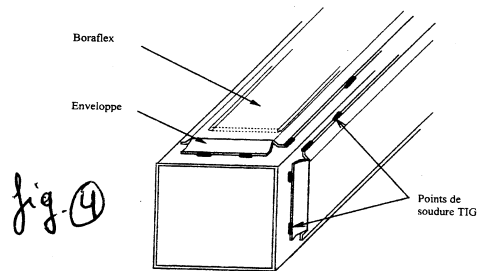
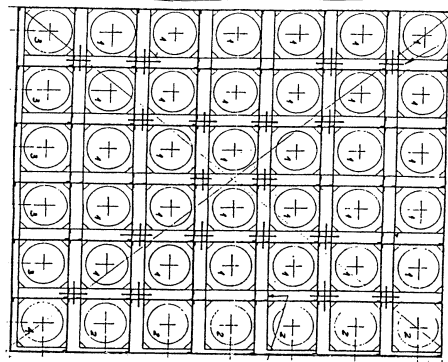
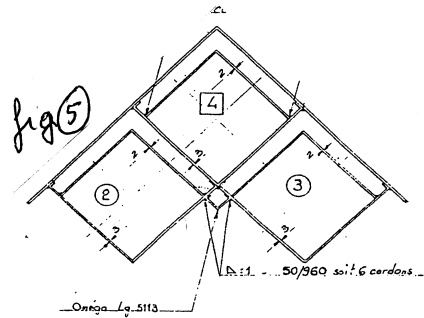


Figure 8 : principe de montage du Boraflex sur une cellule d'un râtelier



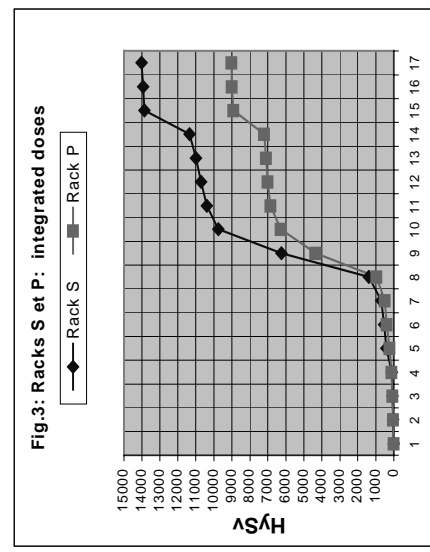
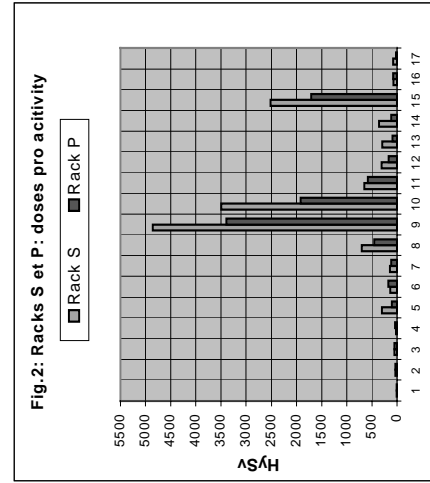
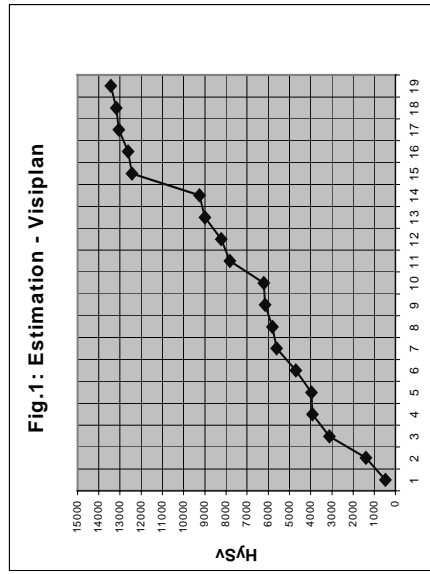
VISPLAN: estimation of the doses

N° Activity	Dose Tot. M.µSv (Fig.1)
1 US decontamination & preparation	469
2 Transfert of the rack to the deconta. unit	934
3 Dimensional contrôle	1718
4 HP decontamination	788
5 Moving the HP cleaning cover	47
6 Putting the rack into the tipper	746
7 Rotation and chipping high and low spot welds	904
8 Putting the rack into the decontamination pool	191
9 Removing the external jackets	361
10 Rotation of 90°	53
11 Cutting and removing the internal jackets	1600
12 HP cleaning	412
13 Moving the HP cleaning cover	778
14 Putting the rack into the tipper	246
15 Rotation and insertion of the BSS plates	3179
16 Transfert of the rack to the deconta. unit	198
17 Dimensional contrôle with dummy	424
18 Replacing the rack in the pool	119
19 Screwing the rack to the pool	252
Total	13419

Dosimetry of the first two racks

N° Activity	Time of activity Hour		Max. indiv. Dose Man.µSv		Total dose Man.µSv (Figure.4)		Total dose integrated (Figure.3)	
	S	P	S	P	S	P	S	P
1 ITV	1	1	0	3	0	6	0	6
2 US	5	5	0	15	0	30	0	36
3 Suction	8	7	-1	24	0	48	0	84
4 Unscrewing	2	8	6	10	22	20	40	104
5 Transfert of the rack to the deconta. unit	8	8	0	61	-18	303	101	-202
6 HP decontamination	24	13	-11	64	88	138	171	33
7 Decontamination of external surfaces	4	4	0	50	6	140	112	-28
8 Removal of the surrounding jackets	14	14	0	204	132	699	451	-248
9 Preparing the rack for cutting	110	47	-63	1349	940	4857	3393	-1464
10 Cutting with holder	195	108	-87	1247	650	3499	1916	-1583
11 Uniform cutting	33	17	-16	249	202	652	577	-75
12 Chemical cleaning	32	4	-28	111	65	307	163	-144
13 HP cleaning	26	7	-19	101	35	290	94	-196
14 Visual ITV inspection	12	7	-5	167	40	356	110	-246
15 Insertion of the BSS plates	77	48	-29	479	398	2510	1709	-801
16 Insertion of the dummy into the cells	4	6	2	22	31	72	79	7
17 Screwing the rack to the pool	4	5	1	24	7	80	21	-59
Total	559	309	-250			14007	9021	-4986

Annex 2



ANALYSIS OF THE DOSES ASSOCIATED WITH THE SPENT FUEL SHIPMENTS FROM THE FRENCH NPPS: ARE THEY ALARA?

J.P. Degrange*, **C. Benhamou***, **L. D'Ascenzo***,
A. Leroy**, **E. Vrignaud*****

* Centre d'étude sur l'Évaluation de la protection dans le domaine nucléaire, France

** Électricité de France GPR, France

*** Électricité de France DC, France

Introduction

The spent fuel shipments from the French Nuclear Power Plants to the La Hague reprocessing plant have always been subject to significant efforts devoted to the prevention and elimination of the non-fixed contamination of the flask surface during their preparation (loading and cleaning) and the workers involved in those tasks receive annual individual dose levels that may be significantly higher than the average. Moreover, these shipments are subject, since 1998, to a reinforced procedure of monitoring of the contamination.

For these reasons, the French electricity utility Electricité de France (EDF) has initiated several studies, performed with CEPN, aiming to obtain a better knowledge of the individual and collective doses associated with the preparation and the monitoring of these shipments:

- A statistical study (from June 1999 to June 2000) has analysed the dosimetric results of the survey of one hundred shipments to La Hague.
- Two extensive measurement campaigns (in 1999 and 2000) have been conducted in order to evaluate the duration and dose-rates associated with each elementary operation involved in the preparation and the monitoring before shipment of a reference flask.

These studies allowed:

- To gain a better knowledge of the distribution of the gamma and neutron doses received during the cask preparation and the contamination monitoring operations, as well as the influence on these doses of the reactor model (900 MWe/1 300 MWe), the fuel type (UO₂/MOX), and the thermal residual power of the assemblies.
- To determine the relative contributions of the main operations, irradiation sources and workplaces to the collective dose, with a specific focus on the operations associated with the prevention, elimination and monitoring of the contamination.
- To identify a set of radiological protection options and past experience analysis that could be envisaged in order to reduce as low as reasonably achievable (ALARA) the collective dose associated with the preparation and monitoring of the spent fuel tasks before their shipment from the NPPs.

Finally, the results of these studies have been used within the framework of a study conducted for the European Commission DG TREN [1], [2].

Statistical study of the French shipments dosimetry

Collective dose variability

As may be seen in Table 1, the statistical distributions of the gamma and neutron collective doses received during the preparation of the cask before shipment (empty cask preparation, fuel loading, cask internal cavity emptying, drainage and drying, decontamination of the cask external surface, cask handling) and the final monitoring against contamination (in the fuel building, on the lorry and on the railway wagon) presented an important variability within the sample under study.

Table 1. **Statistical distribution of the collective dose per shipment (all shipment types taken together)**

Variable		Average (man·mSv)	Standard deviation (man·mSv)	Max (man·mSv)
Preparation	gamma	4.0	2.1	9.9
	neutron	1.2	1.0	6.0
	Total	5.2	–	–
Monitoring	gamma	0.5	0.2	1.2
	neutron	0.8	0.5	2.3
	Total	1.3	–	–
Total		6.5	–	–

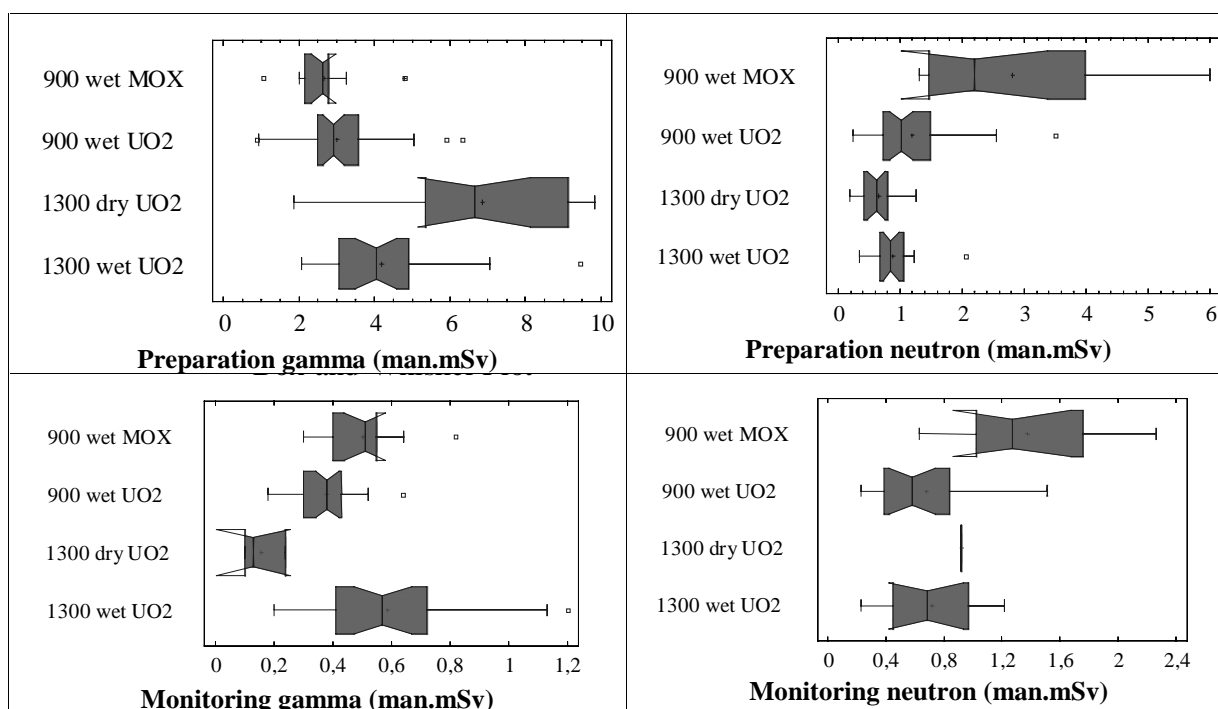
– Not estimated (not enough data).

Influence of the shipment characteristics

As a result of the statistical analysis of the data, a part of this variability could be explained by the influence of the shipment type¹: reactor model (900 MWe/1 300 MWe), loading type (dry/wet) and fuel type (UO₂/MOX), as shown in Figure 1.

1. In some reactors of the 1 300 MWe model, the casks are loaded under water in a loading pit (wet loading) while in some others, they are loaded with no contact of the cask surface with water (dry loading); a “UO₂” shipment involves 12 UO₂ assemblies, while a “MOX” shipment involves only 4 MOX assemblies, surrounded by 8 UO₂ assemblies.

Figure 1. Statistical distribution of the collective dose per shipment, by shipment type (man.mSv)



The analysis of the influence of the shipment type has put in evidence that:

- the gamma collective dose received during the preparation of the cask was the highest in the 1300 MWe reactors (especially for those using the dry loading) and that the neutron collective dose received during either the preparation or monitoring of the casks was significantly higher for the shipments involving MOX fuel;
- in contrast, nor the type of protection used before loading (vinyl cover/3M adhesive) nor the type of package (TN12/LK100) did show any significant influence on the dosimetry.

However, the collective dose for each shipment type presents an important residual variability from one shipment to the other.

Influence of the thermal residual power of the assemblies

As a result of a regression analysis, a part of the residual variability of the collective dosimetry within each shipment type could be explained by the influence of the variability of the residual thermal power of the irradiated fuels – total residual power for the UO₂ shipments, residual power of the UO₂ or MOX assemblies for the MOX shipments – taken here as a first indicator of the variability among the shipments of the dose rates around the cask.

If the influence of the residual thermal power on the neutron collective dose was found significant for most of the shipment types (6 out of 8), it was almost never the case (only 2 out of 8) for the gamma collective dose. This observation has been found coherent with the fact that the contribution to the gamma collective dose of irradiation sources external to the cask (coming from

contaminated tools and/or from the liquid/vapour separator used for the drainage and drying of the cask internal cavity) is significant during the monitoring against contamination and may be predominant during the preparation of the shipment.

However, the influence of the thermal residual power does not explain usually more than the half part of the variability of the collective dose from one shipment to the other, for a given shipment type. Even after taking into account the sources of variability presented above, the residual variability of the collective dose from one shipment to the other remains important and is probably due to the influence of other parameters – not studied here – such as the site, the way of performing the tasks (different from one operator to the other) as well as the possible occurrence of mishaps during some shipments.

Expected collective dose per shipment type and thermal residual power range

This statistical study has allowed to estimate, for each phase of the shipment (preparation, monitoring) and radiation type (gamma, neutron), as well as for each shipment type (reactor model, loading type and fuel type) and – if pertinent – level of thermal residual power:

- the expected average value of the collective dose per shipment; this value could be used to derive dosimetric objectives depending on the shipment characteristics;
- the limits of the 60% statistical “tolerance interval” for the collective dose per shipment (i.e. the intervals inside which one expects to find – with a 95% probability – 60% of the collective dose values for a shipment of a given type); these values could be used to identify and study particularly the shipments that present dosimetric results significantly distant from other shipments of same type (and thermal residual power range, if pertinent), i.e. the shipments that are the most and the least efficient from the dosimetric point of view.

Figure 2 presents the results of such an estimation for the shipment of UO₂ fuel from a 900 MWe reactor, where: the expected average value is given by the central regression line; the upper and lower limits of the 60% “tolerance interval” are given by the two other curves; the three shipments that are the most and least efficient from the dosimetric point of view are indicated by an arrow.

Dosimetric stakes of the preparation and monitoring of the spent fuel shipments

National annual collective dose

The simple multiplication of the average collective dose per shipment by 200 shipments per year has led to a crude estimate of the annual collective dose associated with the preparation and monitoring against contamination of the irradiated fuel casks before shipment, shown on Table 2.

One should note that the resulting annual collective dose of 1.3 man·Sv is not negligible for it corresponds broadly to the annual collective dose associated with the operation and maintenance of one average French reactor and to a little less than 2% of the annual collective dose associated with the operation and maintenance of all French reactors.

Figure 2. Linear regression between monitoring neutron collective dose and residual thermal power (900 MWe reactor, UO₂ fuel shipment)

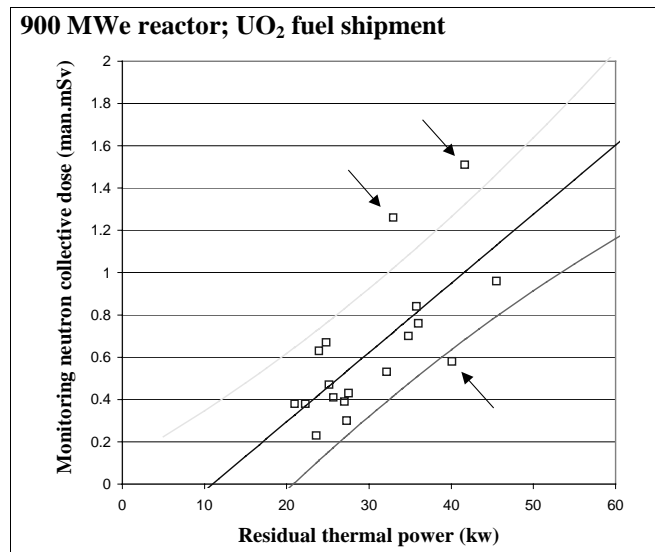


Table 2. Estimate of the national annual collective dose associated with the preparation and monitoring against contamination of the spent fuel casks before shipment

		Collective dose (man·Sv/y)
Preparation	gamma	0.8
	neutron	0.2
	Total	1.0
Monitoring	gamma	0.1
	neutron	0.2
	Total	0.3
Total		1.3

Annual individual dose

The analysis over 10 months of the individual doses of the Chinon B operators has shown that the monitoring against contamination of the spent fuel shipments represents only a small fraction (0.5 mSv/y out of 3 mSv/y, i.e. 16%) of the annual individual dose of the radiation protection department operators involved at least once in the year in the monitoring of a spent fuel shipment, while the preparation of these shipments represents almost the half part (2.5 mSv/y out of 5.7 mSv/y,

i.e. 44%) of the annual individual dose of the general services operators involved at least once in the year in the preparation of a spent fuel shipment.

This analysis has also shown that the difference between the total annual individual dose of the general services operators and the radiation protection department operators (2.7 mSv/y) is due to a great extent to their participation in the preparation of the shipments.

Analytical dosimetric study of the preparation and monitoring of a spent fuel shipment

This study had as an objective the analytical assessment of the dosimetric cost associated with the operations of preparation of the flask and final monitoring of the contamination in order to identify potential protection actions. This assessment has been performed on the basis of measurement campaigns – operation per operation – of the equivalent dose rates and working duration.

Preparation

Description

The operations associated with the preparation of a flask before shipment extend over several days and include the following steps:

- **Tools transfer:** The control module (Figure 4 A) necessary to the emptying, drainage and drying of the flask together with some hoses and connection tools, are moved from one fuel building to the other, monitored against contamination and, if necessary, decontaminated.
- **Package reception:** The empty flask is received and checked against contamination at level 0m of the fuel building. The package is then hoisted towards the preparation pit, prepared for loading and protected against contamination by a water-filled protective skirt above the fins, adhesives on the rear and front end of the cask (Figure 3A), and placed inside a vinyl cover (Figure 3B).
- **Loading:** The package is placed under water into the loading pit (Figure 4G) and loaded with the spent fuel assemblies. The assemblies are then carefully monitored and the cask is closed.
- **Preparation before shipment:** The cask – loaded with the assemblies and filled with pond water – is moved from the loading pit to the preparation pit and its protections (vinyl cover and protective adhesives) are unfitted (Figures 3C and 3D). After a rapid decontamination of the cask, the preparation of the cask before shipment (as such) starts and will extend over several days. As the French casks do not allow, for safety reasons, transportation of the spent fuel assemblies without the complete elimination of the inside pond water, the cask must be emptied, drained and dried carefully, with the help of a control module (mobile module containing the pumps, valves, as well as measurement and display devices, Figure 4A), a liquid/vapour separator (Figure 4D), as well as several connecting tools and hoses (Figures 4B, 4C, 4D). It is then subject to extensive leak tightness monitoring (Figure 4E), at the end of which the skirt is emptied and removed and protective adhesives put in place (Figure 3G). The cask is then taken

down to the 0m level of the fuel building and ready for shipment, after the putting in place of the shock absorbers on its front and rear ends.

- Decontamination: Decontamination of the whole cask surface (Figures 3E and 3F) takes place at least once in a shift during all the preparation operations.
- Monitoring at the railway station: Package is shipped after the end of monitoring against contamination at the railway station (Figure 4H).

Contribution of preparation steps to the collective dose

Table 3, that presents the working duration, average dose rate and collective dose associated with each of the preparation steps, shows clearly that the operations dealing with the preparation before shipment and the decontamination of the cask, that represent less than the half part (47%) of the working duration, contribute to the most part (85%) of the collective dose, due to the significant average dose rate to which the workers are exposed during these operations.

Table 3. **Collective dose associated with the various steps of the flask preparation (900 MWe, UO₂)**

Operation	Duration	Duration	Dose	Average dose rate		
	(h)	%	%	(μSv/h)		
			Gamma + neutron	Gamma	Neutron	Gamma + neutron
Tools transfer	25.0	16	2	2	0	2
Package reception	30.9	20	6	7	0	7
Loading	26.9	17	7	8	0	8
Preparation before shipment	68.6	44	74	19	15	34
<i>Decontamination</i>	4.5	3	11	34	44	78
Monitoring at railway terminal	0.2	0.1	0.0	0	0	0
Others	0.4	0.2	0.1	3	2	5
Total	156.4	100	100	12	8	20

Contribution of working areas to the collective dose

The working areas that contribute the most to the collective dose are: the working areas close to the cask (51% of the collective dose), mainly for the relatively high dose rate present in these areas, coming as well from gamma and neutron radiation; the circulation areas and working areas back from the cask (34% of the collective dose) for which the exposure comes essentially from the large time spent in these areas and where the dose rate results mainly from the gamma radiation; the other zones (15% of the collective dose) whose contribution comes also essentially from the large time spent in these areas and where the dose rate results also mainly from the gamma radiation.

Contribution of the most important operations to the collective dose

The operations that contribute mostly to the collective dose are, on the one hand, the operations associated with the prevention and elimination of the contamination and, on the other hand, those associated with the flask preparation, that contribute respectively to 29% and 34% of the collective dose.

Regarding the collective dose associated with the prevention and elimination of the contamination: 11% comes from the decontamination operations, of which the most part (90%) is due to the decontamination operations performed after emptying the internal cavity of the cask, during which the average dose rates are much higher (respectively 2 and 7 times higher for gamma and neutron) than the dose rates at the same locations before emptying the cavity; 10% comes from the fitting and unfitting of the adhesives, of which almost 90% is due to the fitting of ill-suited cut adhesives during the followed up campaign; 8% come from the fitting, rinsing, and unfitting of the vinyl cover, of which the most part (80%) is due to the unfitting of the cover after loading.

Regarding the collective dose associated with the flask preparation, 14% comes from the flask tightness monitoring, of which a third is due to the gamma radiation from the contamination of the draining tool; 8% comes from the emptying and removal of the skirt, of which almost the half part (43%) is due to the gamma radiation coming from the contamination of the liquid/vapour separator during the rinsing of the later with the hot water drained from the skirt; 6% comes from the draining and drying of the internal cavity, of which almost three quarter (73%) is due to the gamma radiation from the separator and the draining tool that are permanently present during the drying of the internal cavity, that represents 64% of the collective dose of these operations; 6% comes from waiting times and forms filling, whose contribution to the collective dose is mainly due to their important duration that represents 20% of the total duration of operations, and of which more than 80% are received at a desk, located a few meters away from the cask inside the fuel building.

A specific attention has been devoted to the operations concerning the liquid/vapour separator, that contribute to 6% of the collective dose, of which 44% is received during the rinsing of the separator while 20% come from the monitoring and emptying of the separator during the draining of the cavity.

Possible protection actions

On the basis of the analysis of the contributions presented above, a set of possible protection actions has been identified and their potential dosimetric savings have been evaluated: liquid/vapour separator shielding (<11%), decontamination only with full internal cavity (<7%), protecting the desk from radiation (5%), remote display devices in the protected desk (2%), elimination of special adhesives for protection and transport (10%) or use of well-suited adhesives for transport (5%), removal of the liquid/vapour from the draining orifice besides which a lot of operations take place (4%), forms filling outside the fuel building (3%), early exit of the operators at the end of operations (2.5%), remote monitoring and emptying of the liquid/vapour separator (1%), systematic decontamination of the immersion tool (<0.5%).

The maximal potential dosimetric savings associated with the implementation of all these protection actions (among which some are mutually exclusive) has been estimated to 40% of the collective dose associated with the preparation of the cask before shipment.

Monitoring against contamination

Description

The monitoring of the cask's surface against contamination begins after completion of the cask's preparation and includes several monitoring steps: in the preparation pit of the fuel building, on the lorry and on the railway wagon (the second step being omitted when the NPP is directly connected to the railway network).

Monitoring in each step consists of both one screening test for each location defined in the harmonised procedure (covering the totality of the accessible zones of the cask, with the exception of the cooling fins, thousands in number) and one 300 cm² smear test for each of the same locations. Monitoring on the lorry and at the railway station include as well screening tests and smear tests of the accessible parts of the vehicle. All these tests are successively performed by two teams: one from the NPP radiation protection department and one from an independent organisation (double monitoring). The total number of 300 cm² smear tests performed for each shipment is thus equal to 422.

Finally, regulatory dose-rate measurements (at contact and 1-meter away from the cask) are performed two times: once before the shipment of the lorry and once before the shipment of the railway wagon.

Contribution of the monitoring steps to the collective dose

The study of the relative contribution to the collective dose of each step of (double) monitoring against contamination, has shown that the contribution to the collective dose of the first step (in the fuel building) is almost equal to the double of each of the two other steps (on the lorry and on the railway wagon). This difference is essentially due to the difference in duration between these steps, that does not come only from the number of smear tests but also from the sequence of operation, waiting and handling phases within each step.

Contribution of operations to the collective dose

The study of the relative contribution to the collective dose (all monitoring steps together) of the various monitoring operations has shown that 90% of the collective dose was associated with the monitoring operations as such (front part (29%); rear part (30%); waiting /circulation (21% of the dose for 63% of the duration); Fins zone (7%); lorry and wagon (4%) and that only 10% of the collective dose were received during operations that were not directly related to contamination (regulatory dose rate measurements (7%) and seals affixing (3%)).

Monitoring dose and residual contamination risk by monitoring zone

The comparison, for each monitoring zone, of the collective dose associated with the monitoring and the residual risk of finding contamination (estimated on the basis of the contamination incidents detected in 1997 at Valognes railway terminal, on casks coming from French NPPs) has allowed to sort the different cask monitoring zones according to a "monitoring interest index" calculated as the relative probability of finding some residual contamination after the monitoring on

that specific zone compared to the other zones, divided by the collective dose associated with the monitoring of that zone against contamination.

This index may be used as a “dosimetric cost – monitoring effectiveness” criterion that may allow to separate the zones for which a reinforced monitoring seems particularly interesting from those for which a less stringent monitoring could be envisaged.

This comparison has led to divide the cask into three groups: trunnions and trunnion bases, vertical and oblique parts of the front end, skirt side and seal face of the front part, for which the relative probability (compared to the other zones) of residual contamination reaches 80% while the monitoring of these zones represent only 36% of the collective dose associated with monitoring operations; vertical and oblique parts of the rear end, skirt side and seal face of the rear part, shock absorber at the rear end, with a relative probability of residual contamination of 20.5% and a contribution of 36% to the collective dose; horizontal parts of the front and rear ends, shock absorber at the front end, cooling fins, with a null probability of residual contamination in 1997 and a contribution to the collective dose that reaches 28% (11% for the cooling fins).

Possible protection actions

On the basis of the analysis of the contributions presented above, a set of possible protection actions has been identified and their potential dosimetric savings has been evaluated: reduction of waiting time/removal from cask (10.5%); discontinuation of double monitoring for the zones with no contamination detected in 1997 (10.5%).

The maximal potential dosimetric savings associated with the implementation of all these actions has thus been estimated to 21% of the collective dose resulting from the monitoring against contamination of the cask before shipment.

Conclusions

The estimate of the annual collective dose associated with the preparation and monitoring against contamination of the irradiated fuel casks before shipment from the French NPPs (1.3 man·Sv/y) is not negligible and corresponds broadly to the annual collective dose associated with the operation and maintenance of one average French reactor as well as a little less than 2% of the annual collective dose associated with the operation and maintenance of all French reactors.

Moreover, the preparation of the casks before shipment may represent almost the half part of the annual individual dose of the members of the general services involved at least once in the year in the preparation of a spent fuel shipment and may explain to a great extent the difference between the annual individual dose of the general services operators (involved in the preparation) and the radiation protection department operators (involved in the monitoring).

The operations of prevention, elimination and monitoring of the surface contamination of the irradiated fuel casks before shipment contribute significantly to the collective dose received by the operators during the operations of preparation and monitoring of the cask before shipment: 29% of the preparation collective dose and 90% of the monitoring collective dose, i.e. 42% of the total collective dosimetry.

Identified protection actions present potential dosimetric savings that are non negligible but limited: these potential savings are limited to 40% of the preparation collective dosimetry, and 21% of the monitoring collective dosimetry, i.e. 36% of the total collective dosimetry.

As shown in Table 4, protection actions directly associated with the monitoring of the cask against contamination present much more limited potential savings (4% of the total collective dose) than protection actions associated with the preparation of the flask before shipment (31% of the total collective dose). However, these savings could reach more than 11% of the total collective dose if double monitoring was discontinued for every monitoring zone of the cask.

Table 4. **Detailed potential dosimetric savings of proposed protection actions**

Protection action	Dosimetric saving Preparation + monitoring collective dose %
Liquid/vapour separator shielding	< 9*
Elimination of special adhesives (before loading and transport)	8*
Decontamination operations with full cavity	< 6*
Desk protected from radiation	4*
Remote display devices in the protected desk	2*
Special adhesives (before transport) well suited	4
Removal of the liquid/vapour separator from the draining orifice	3*
Forms filling outside the fuel building	2
Early exit of operators at the end of operation	2
<i>Reduction of waiting time / removal from cask</i>	2*
<i>Discontinuation of double monitoring for fins zone</i>	1*
<i>Discontinuation of double monitoring for caisson horizontal part</i>	1*
Remote monitoring/emptying of the liquid/vapour separator	1
Total preparation	31
Total monitoring	4
Total preparation + monitoring	36

* Actions considered in the sum.

Results associated with monitoring operations are presented under italics.

Other protection actions, whose potential interest has not been (entirely) quantified, could be envisaged : choice of the best methods of protection of the cask external surface against contamination by the loading pond water; minimisation of operation mishaps and cask contamination events during the preparation before shipment; optimisation of each step of the monitoring procedure in terms of

number/location of the smear tests and interest of a double control in order to minimise the collective dose spent to reach a same level of risk of residual contamination.

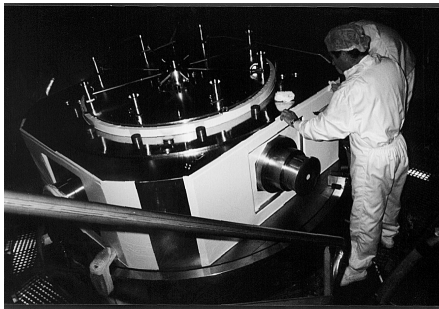
The implementation of these actions would require the organisation of past experience collection and analysis regarding: the respective interest of the different protection methods (vinyl cover, adhesives, ...) in terms of both contamination prevention effectiveness and dosimetric impact on the preparation operations; the occurrence frequency and causes of operation mishaps and cask contamination events; the frequency and location of contamination events detected during the preparation of the cask as well as during its monitoring and double monitoring.

Finally, the studies described above have demonstrated the feasibility as well as the interest: of extensive analytical studies of the dosimetry – on a task per task basis – to identify the most potentially effective protection actions; of detailed statistical analysis of the dosimetry for the definition of dosimetric objectives (on the basis of expected average values) and the optimisation of the past experience collection (with the help of statistical “tolerance intervals” aiming to identify the good practices as well as the operation mishaps that may lead to an excessive dosimetry); in order to keep as low as reasonably achievable (ALARA) the doses associated with the spent fuel shipments from the French NPPs.

References

- [1] Hughes J.S., Shaw R.B., Gelder R., Schwarz G., Fett H.J., François Y., Dello N., Desnoyers B., Tchatalian B., Van Hienen J., Jansma R., Lefaire C., Degrange J.P. Application of the ALARA principle to the decontamination of transport of irradiated fuel. Joint Report NRPB/GRS/TRANSNUCLEAIRE/NRG/CEPN, June 2001.
- [2] Hughes J.S., Shaw R.B., Gelder R., Schwarz G., Fett H.J., François Y., Dello N., Desnoyers B., Tchatalian B., Van Hienen J., Jansma R., Lefaire C., Degrange J.P. – Applying ALARA to the decontamination of irradiated Nuclear Fuel Containers. In: “Packaging and Transportation of Radioactive Materials” (PATRAM ‘01), Proceedings of the 13th Conference, Chicago, USA, 3-7 September 2001.

Figure 3. Operations of prevention and elimination of contamination



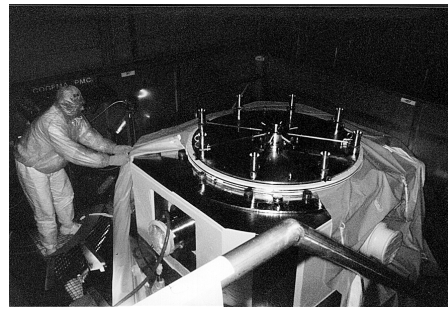
A: Adhesives fitting before loading



B: Cover fitting before loading



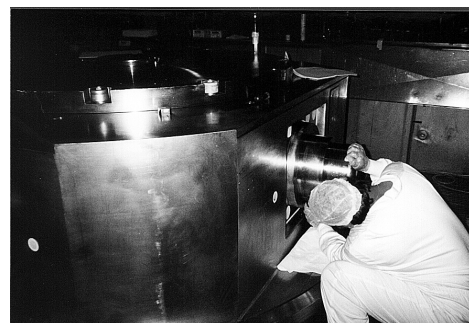
G: Adhesives fitting before shipment



C: Cover unfitting after loading



D: Adhesives unfitting after loading

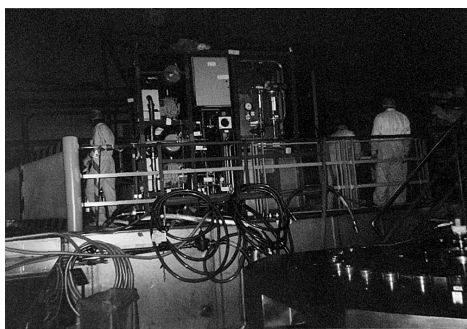


E: Decontamination of front part

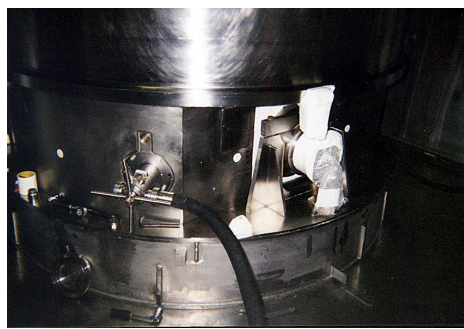


F: Decontamination of rear part

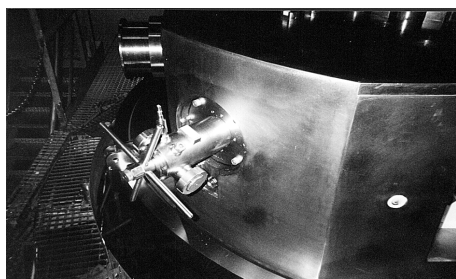
Figure 4. Flask preparation



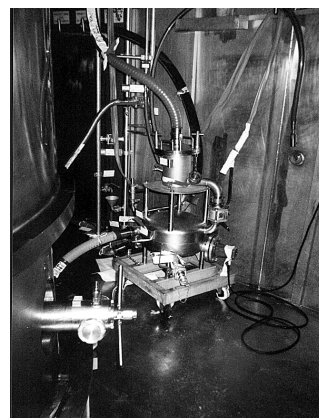
**A: Emptying/draining/drying
(control module)**



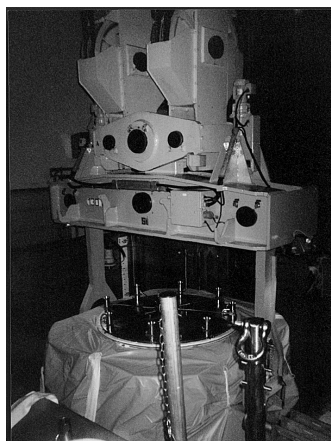
**C: Emptying/draining/drying
(B tool)**



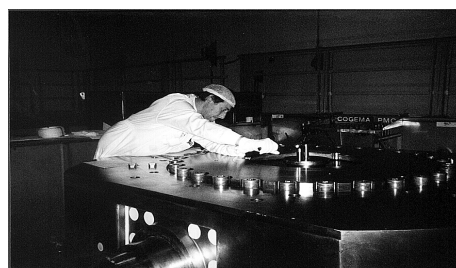
**B: Emptying/draining/drying
(A tool)**



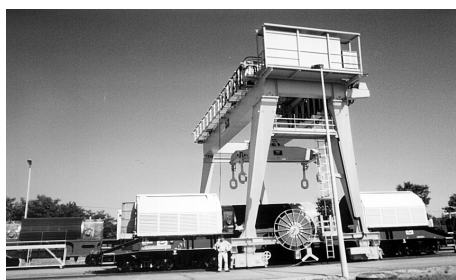
**D: Emptying/draining/drying
(B tool, liquid/vapour separator)**



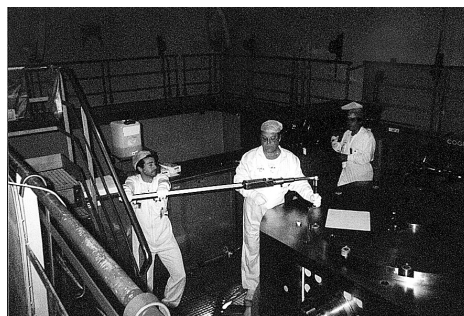
G: Handling (fuel building)



E: Tightness monitoring



H: Handling (railway terminal)



F: Other operations (tightening)

DOSES OF THE STAFF DURING THE SPENT FUEL ASSEMBLIES TRANSPORTATION AND STORAGE IN NUHMOS 56V CONCRETE SYSTEM

V. Atoyán, A. Muradyán
Armenian NPP

The NUHMOS 56V concrete system provides long-term interim storage (50 years) for spent fuel assemblies, which have been out of the reactor for a sufficient period of time. It consists from horizontal storage modules (see Figure 1). The fuel assemblies are confined in a helium atmosphere by a canister containment pressure vessel (see Figure 2). The canister is protected and shielded by a massive reinforced concrete module. Decay heat is removed from the canister and concrete module by a passive natural draft convection ventilation system.

The project of storage does not foresee the radiation monitoring inside of building and around it. But we provided and realise the radiation monitoring program around storage, it includes tree phases:

- determination the zero background around the building before storage put in exploiting;
- monitoring of the radioactive particles in air (additional aspiration plant); dose rate monitoring by portable dosimeters and soil monitoring during the process of the fuel storage;
- constantly after the completion the fuel storage process – monitoring of the radioactive particles in air (additional aspiration plant); dose rate monitoring by portable dosimeters, and soil monitoring. Also designed the dose rate monitoring by the dosimeter RME³ with the transfer of data by radio channel to central monitor.

The canisterised spent fuel assemblies are transferred from the plant's spent fuel pool to the concrete storage modules in a transfer cask (see Fig. 3). The cask is aligned with the storage module and the canister and inserted into the module by means of a hydraulic ram. The system is a totally passive installation that is designed to provide shielding and safe confinement of spent fuel for a range of postulated accident conditions and natural phenomena.

The primary operations for system are:

1. Cask preparation

Cask washdown and interior decontamination.

2. The dry shielded canister (DSC) preparation

The internals and externals of the dry shielded canister are washed or wiped down.

3. Placement DSC in cask

The empty dry shielded canister placed into the transfer cask. Proper alignment is assured by visual inspection.

4. Fill with water and seal cask/DSC annulus

The transfer cask and dry shielded canister inside the cask are filled with water. This prevents an in-rush of pool water as they are placed in the pool. The dry shielded canister/cask annulus is sealed prior to placement in the pool. This prevents the contamination of the dry shielded canister outer surface by the pool water.

5. Cask lifting and placement in pool

The water-filled transfer cask, with dry shielded canister inside, is then lifted into the fuel pool and positioned in the cask laydown area.

6. DSP spent fuel loading

Spent fuel assemblies are placed into the dry shielded canister basket. This operation is identical to the presently used at plant for shipping cask loading.

7. DSC top shield plug placement

This operation consists of placing the dry shielded canister top shield plug onto the dry shielded canister using the plant's crane.

8. Lifting cask from pool

The loaded cask is lifted out of the pool and placed (in vertical position) on the decontamination area. This operation is similar to that used for shipping cask handling operation.

9. Decontamination of the cask

The loaded cask is decontaminated and then transferred to the work cask area.

10. Inner DSC sealing

Using a pump, the water in the dry shielded canister is lowered below the inside surface of the dry shielded canister top shield plug. The inner top cover is put in place and a seal weld is made between the edge of the cover plate and the dry shielded canister shell. This weld provides the inner seal for the dry shielded canister.

11. DSC drying and backfilling

The initial blow-down on the dry shielded canister is accomplished by pressurising the vent port with neutral gas. The water in dry shielded canister is forced out and goes back to the fuel pool. The dry shielded canister then evacuated to remove the residual liquid water and water vapor in the dry shielded canister cavity. When the system pressure has stabilised, dry shielded canister is backfilled with helium and re-evacuated. The second backfill and

evacuation ensures that reactive gases remaining are less than 0,25% by volume. After the second evacuation, the dry shielded canister is again backfilled with helium and slightly pressurised. A helium leak test of the inner seal performed. The helium pressure is reduced, and the drain and fill port penetrations seal welded closed.

12. Outer DSC sealing

After Helium backfilling, DSC outer top cover plate is installed by placing a second seal weld between the cover plate and the DSC shell Together with the inner seal weld, this weld provides a redundant seal at the upper end of DSC. The lower end has redundant seal welds, which are installed and tested during fabrication.

13. Cask/DSC annulus draining and top cover plate placement

The transfer cask is drained, removing the demineralized water from the cask/DSC annulus. A swipe is then taken over the DSC exterior at the DSC top cover plate and the upper portion of the DSC shell. Clean demineralised water is finished through the cask/DSC annulus to remove of any contamination left on the DSC exterior as required. Then the transfer cask top cover plate is put in place using crane. The cask lid bolted for subsequent handling operations.

14. Transport the loaded cask to HSM

Upon entering the ISFSI secured area, the transfer cask is positioned and aligned with the particular HSM in which a DSC is to be transferred.

15. Cask/HSM preparation

At the ISFSI with the transfer cask positioned in front of the HSM, the cask top cover plate is removed. The trailer is backed into close proximity with the HSM and HSM door is removed. The skid positioning system is used for final alignment and docking of the cask with the HSM.

16. Loading DSC into the HSM

After final alignment of the transfer cask HSM and hydraulic rum the DSC is pushed into the HSM by the hydraulic carriage.

17. Storage

After the DSC is inside the HSM, the hydraulic carriage is disengaged from DSC and withdrawn through the cask. The trailer is pulled away, the HSM shielded access door installed. The DSC is now in safe storage within the HSM.

In framework of the project were designed dose rates at the operations by unloading and transporting the fuel assemblies, they are shown in Table 1.

Using data of Table 1, we calculate the doses at the carrying-out of operations by loading and transportation the spent fuel assemblies, they are shown in Table 2.

We also want to show you the data, describing the radiation situation during the loading the dry shielded canister (see Table 3).

The measuring of doses of the personnel was carry out by electronic and TLD dosimeters.

Comparison the data of the lust two columns of the Table 2 shows that the doses, calculated by our specialists, are about 1,5 times less then real doses. The main cause of it is that the designed in the project dose rates, shown in Table 1, are found less than real dose rates. The cause of that, was that circumstance, that during the designed calculations were taking into account the doses from the spent fuel assemblies which storage time in spent fuel pool is 3 years, but the real storage time was 10 years. Taking into account the experience obtained by our specialists, we can expect, that in future the similar operations by storing the spent fuel assemblies would bring to more little doses.

In particular, the doses obtained during the operations by: decontamination of the cask and dry shielded canister and placement-replacement of the additional temporary screens are equals 2,51 mSv (11% from total dose) and 7,14 mSv (31% from total dose) accordingly. It is enough much. And we can decrease them with the help of decreasing the time of operations.

Рисунок 1.3-3

Схема горизонтально расположенного модуля хранения NUHOMS®

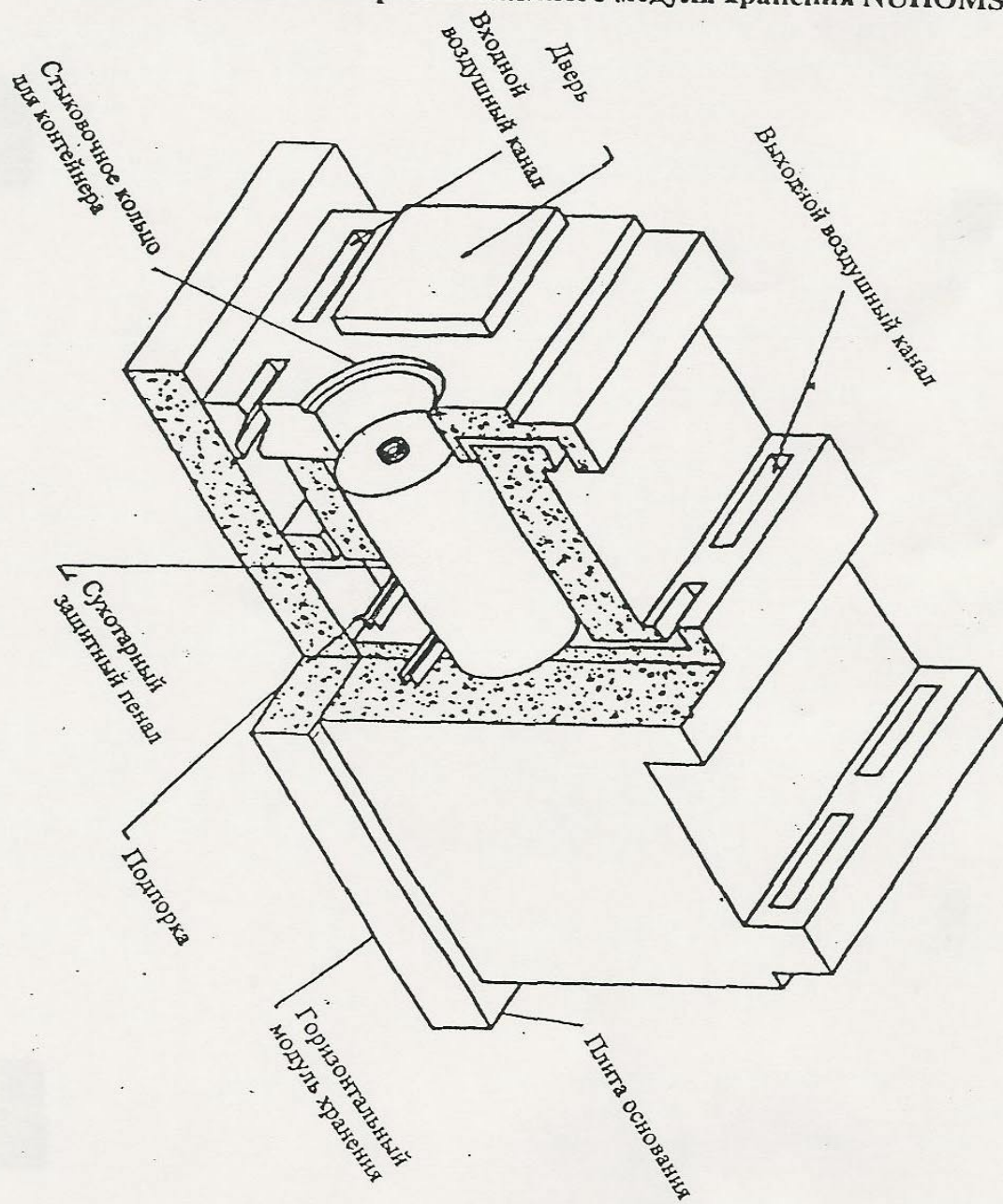


Рис. 1.3-5
 Перегрузочный
 защитный контейнер

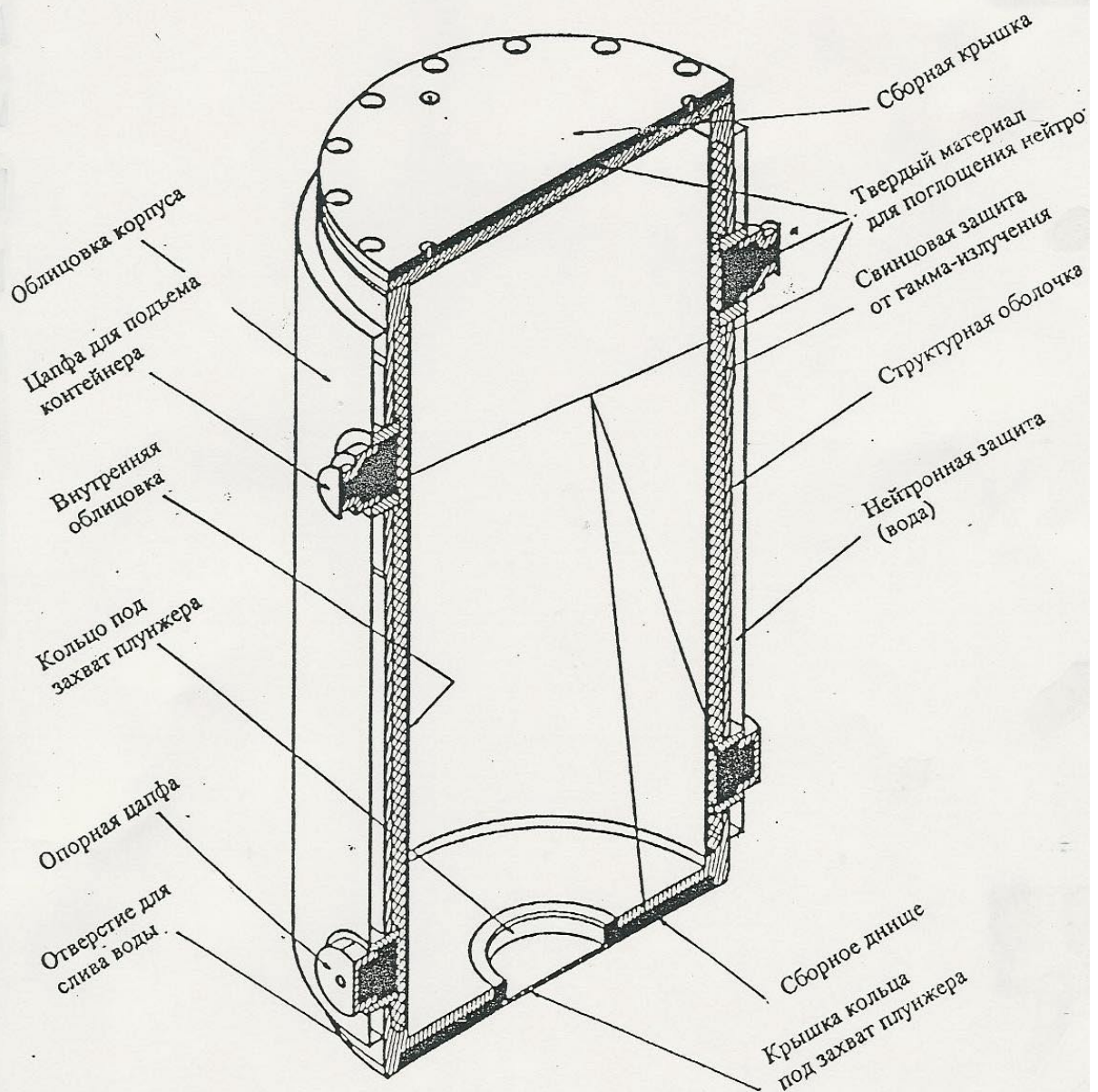


Рисунок 4.2-7

Общий вид перегрузочного защитного контейнера NUHOMS и СЗП с установленными ОТВС

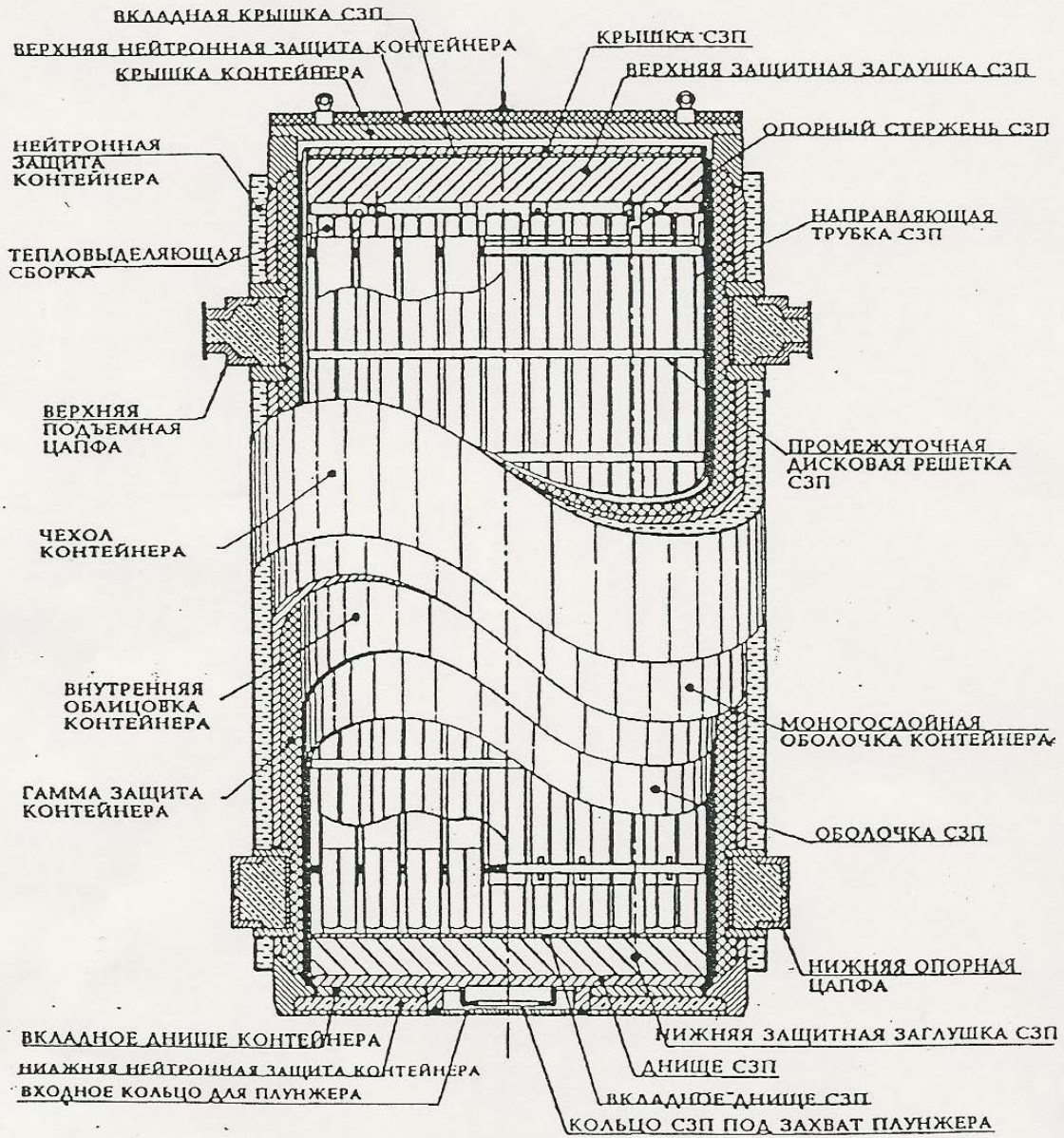


Table 1. The designed dose rates at the operations by unloading and transporting the fuel assemblies

Situation	Direct dose rate [$\mu\text{Sv/h}$]			Total
	Kind of contact	Neutron	gamma	
1. Dry shielded canister during the pressurising: 1.1 a) the canister is fill of boron water; b) the water is between the DSC and cask; c) with the install upper cover.	Axial contact 1 meter	88 85	1093 837	1181 922
	Radial contact 1 meter	341 159	2240 1100	2581 1259
	Axial contact 1 meter	320 308	1847 1390	2167 1698
1.2 The same after moving off the little volume of water a) from cavity of the dry shielded canister; b) from ring clearance between dry shielded canister and cask.	Axial contact 1 meter	272 261	826 637	1098 898
1.3 The same after installs the cover.	Axial contact 1 meter	30 29	169 129	199 158
1.4 The same after installation the temporary protection Screen.	The upper edge	24	135	159
1.5 The same without the boron water spent fuel assemblies.	Axial contact 1 meter	510 484	408 306	918 790
	Radial contact 1 meter	520 250	2700 1280	3230 1530
1.6 The same without temporary protection.	Axial contact 1 meter	6219 5947	2026 1519	8245 7466
1.7 The same after installs the cover.	Axial contact 1 meter	4752 4540	616 470	5368 5010

1.8 The same after installing the temporary protection.	Axial contact 1 meter	377	143	520
		356	110	466
	The outside edge	302	114	416
1.9 The same without the water ring clearance between the dry shielded canister and cask.	Radial contact	710	2858	3568
		330	1363	1693
	1 meter			
1.10 The cover of cask is install.	Axial contact 1 meter	430	25	455
		402	19	421
2. Dry shielded canister in horizontal storage module		1,74	117,4	179,1
The roof of horizontal storage module.	<ul style="list-style-type: none"> the contact dose of direct radiation the dose rate of radiation from the grate (diffused) 	0,35	1,5	1,85
The side wall of horizontal storage module.	<ul style="list-style-type: none"> the contact dose 	0,89	53,7	54,6
The closed door of horizontal storage module.	<ul style="list-style-type: none"> the contact dose 1 meter 	38,2	10,4	48,6
		35,8	6,4	42,2
The open door of horizontal storage module	<ul style="list-style-type: none"> the contact dose 	5622	1119	6741
3. Dry shielded canister during the transportation in the protection cask		710	2858	3568
	radial contact	330	1363	1693
	1 meter			
	contact of the upper axial direction	430	25	455
	1 meter	402	19	421
	contact of the upper axial direction	440	190	630

Table 2. Calculated and real doses at the carrying-out of operations by loading and transportation the spent fuel assemblies

Operation	Number of workers	The time of carrying-out the operation (hours)	The dose rate $\mu\text{Sv/h}$	Calculated dose mSv	Real doses mSv
Loading of dry shielded canister					
Preparation the cask and dry shielded canister	2	4	10,8	0,05	0,04
Placing the cask and dry shielded canister in pool	3	10	10,8	0,10	0,06
Loading the spent fuel in dry shielded canister	3	16	72	1,28	0,83
Placing the upper shield cover of dry shielded canister	3	1	72	0,08	0,06
Extraction the dry shielded canister from the pool and placing it in the decontamination area	5	2	1181	2,40	1,62
Total for operation					
				3,91	2,61
Pressurising the dry shielded canister					
Decontamination of the outside surface of the cask	2	1	2581	2,50	1,68
Discharging the water over the shield cover of dry shielded canister	3	1	1181	1,20	0,74
Decontamination the upper part of the cask and dry shielded canister	2	1	1181	1,20	0,83
Removing the compression from the ring clearance between the cask and dry shielded canister, placing the cover and welding devise	2	2	2167	4,20	2,85
The automatics welding the cover with the body of dry shielded canister	2	4	10,8	0,05	0,04
Placement of the additional temporary screens	2	0,5	199	1,00	0,61

Performance of metal control	2	1,5	199	0,30	0,20
The vacuum drying and pumping the helium in the dry shielded canister	2	16	199	0,80	0,48
Testing the joints for hermetic	2	1	199	0,20	0,12
Welding the covers of ventilation opening and siphon	2	1,5	199	0,30	0,21
Removing of the additional temporary screens	2	0,5	8245	4,10	2,67
Placement of the cover	2	0,5	5368	2,65	1,75
Weld with the welding devise	2	9	10,8	0,10	0,06
Placement of the additional temporary screens	2	0,5	5368	2,65	1,62
Removing the welding devise from the dry shielded canister	2	0,5	520	0,26	0,17
Performance of metal control	2	2	520	1,00	0,60
Removing of the additional temporary screens.	2	0,5	5368	2,65	2,24
Discharging the water from the ring clearance between the cask and dry shielded canister	2	1	455	0,40	0,26
Placement the cover of the cask	2	1	455	0,40	0,26
Total for operation				25,56	17,13

Loading the cask for transportation						
Preparation the supporting carriage of the cask and the trailer	2	2	10,8	0,02	0,010	
Placement the cask on the supporting carriage of the trailer	2	0.5	455	0,20	0,008	
Replacement the bottom and placement the protection cover	2	1	455	0,45	0,30	
Fixing the cask on the carriage	2	1	455	0,45	0,28	
Total for operation				1,12	0,60	
Transportation dry shielded canister to the horizontal storage module						
The horizontal storage module preparation	2	2	0,1-0,2	----	0	
Transportation the cask to storage	6	1	1693	1,60	1,12	
Removing the cover from the cask	3	1	455	0,45	0,26	
Attachment the cask with the horizontal storage module	3	2	455	0,90	0,68	
Placement the hydrocylinder and attachment with the cask	3	1	455	0,45	0,35	
Push out the dry shielded canister from the cask	3	0.5	455	0,22	0,12	
Placement the hydrocylinder on the trailer and un-docking the cask from the horizontal storage module. Placement the seismic limit	3	0.1	6741	0,60	0,24	
Placement the door of the horizontal storage module	3	1	179,1	0,20	0,06	
Total for operation				4,42	2,83	
Total				35,01	23,17	

Table 3. Radiation situation during the loading the dry shielded canister

Operation	Dose rate in different points ($\mu\text{Sv}/\text{sec}$)									
	-	-	-	-	-	-	-	-	-	-
Loading the dry shielded canister in container compartment. The water level in container compartment – 6,7 meters.										
Extraction the dry shielded canister from the container compartment and its decontamination. The water level in container compartment – 10,5 meters. Contamination of the dry shielded canister surface is 360–400 β -particles/ sm^2 minute.	0,10	0,06	0,03	0,02	0,02	0,02	0,02	0,02	0,04	0,05
Installation the first cover	0,08	0,07	0,02	0,02	0,02	0,02	0,02	0,02	0,04	0,05
Pump out the water from the dry shielded canister	0,16	0,10	0,04	0,06	0,07	0,07	0,07	0,07	0,04	0,05
The vacuum drying of dry shielded canister	0,15	0,10	0,04	0,06	0,07	0,07	0,07	0,07	0,04	0,05
Compression the helium in to the dry shielded canister	0,03	0,10	0,05	0,06	0,07	0,06	0,06	0,06	0,04	0,05
Installation the second cover	0,01	0,06	0,03	0,06	0,05	0,06	0,05	0,06	0,04	0,05
Notice: the alpha particles and neutrons does not recorded										

SESSION 3

ENHANCEMENT OF RADIOLOGICAL SAFETY CULTURE WITHIN IGNALINA NPP AND CONTRACTORS

V. Pletniov

Ignalina Nuclear Power Plant, Lithuania

G. Klevinskas

Radiation Protection Centre, Lithuania

Abstract

The Lithuanian Hygiene Standard HN 83:1998 “Radiation Protection of Outside Workers” (1998), that is based on requirements set out in the Law on Radiation Protection of the Republic of Lithuania (No. VIII-1190 of 12 January 1999), [1] and other legal acts, establishes the radiation protection requirements for contractors (hereinafter – outside workers) when they are performing their activities within the controlled areas. The principal requirement of the Lithuanian Hygiene Standard HN 83:1998 is that the radiation protection of outside workers shall be at the same scale as of permanent workers. One of crucial points for ensuring the high radiological safety culture between the license holder, employer and outside workers is the clear allocation of tasks and distribution of responsibilities in the field of radiation protection.

The basic requirements for radiation protection of outside workers and main factors that help to enhance the radiological safety culture within Ignalina NPP and outside workers are discussed in the paper.

Legislation

The radiation protection requirements of outside workers when they are performing their activities within the controlled areas are established by the following legislation:

- Lithuanian Hygiene Standard HN 73:2001 “Basic Standards of Radiation Protection”, (2001).
- Lithuanian Hygiene Standard HN 87:2001 “Radiation Protection in Nuclear Power Plant”, (2001).
- Lithuanian Hygiene Standard HN 83:1998 “Radiation Protection and Safety of Outside Workers”, (1998).
- Government Resolution No. 653 “On Regulations of Licensing the Practices Involving Sources of Ionizing Radiation”, (1999).
- Other legal acts.

The local instructions establishing radiation protection requirements at Ignalina NPP are prepared on the base of above-mentioned regulations.

The principal requirement established by [2] is that the radiation protection of outside workers working within the controlled area of the Ignalina NPP shall be at the same scale as of permanent workers. According to [2], the employers whose workers are performing their activities within the controlled area of the nuclear power plant, shall establish the co-operation agreements with license holders, where the order and procedure of registration and estimation of workers exposure, measures for exposure reduction, the order of allocation of tasks and distribution of responsibilities between employer and license holder and other significant means from the radiation protection point of view shall be described.

The activities of outside organisations within the controlled area are licensed according to [3]. The Radiation Protection Centre is the regulatory body that according to the provisions set out in the Law on Radiation Protection, among other functions, is empowered to issue the licenses to conduct the practices involving sources of ionising radiation or under the influence of ionising radiation. The Radiation Protection Centre carries out periodical radiation protection inspections of outside organisations.

Passbook of outside workers' exposure

The activities within the controlled area of the Ignalina NPP are not allowed without the license and without the Passbook of Outside Worker's Exposure. The Passbooks of Outside Workers' Exposure are issued only for those workers of particular license who are included into the license annex. The exposure results of outside workers, after they are finished work connected with ionising radiation, are recorded to the Passbook of Outside Workers' Exposure. There are a number of advantages for use of the Passbook. For the license holder, the Passbook gives all information about the worker, his medical suitability to work and his exposure results during the previous activities connected with ionizing radiation. After the outside worker has finished his activities connected with ionizing radiation, all information on exposure results is submitted to the State Register of Radiation Sources and Exposure to Workers according to the procedure established by [4]. The form of the Passbook of Outside Worker's Exposure is presented in Figure 1.

Tasks and responsibilities of administration of outside organisation

The administration of outside organisation is responsible for the following measures as regards the assurance of radiation protection of outside workers:

- competent preparation and improvement of qualification of outside workers in the field of radiation protection;
- conducting periodical medical examination of outside workers working within the controlled area of Ignalina NPP;
- training, education and instructing of outside workers in the field of radiation protection;
- registration of exposure results and their presentation to the Radiation Protection Centre according to order established by [4];

Tasks and responsibilities of Ignalina NPP

The Ignalina NPP, as the license holder, controls how internal rules on radiation protection and work safety are being followed by outside workers.

The license holder is also responsible for setting up of the main dosimetric parameters (collective dose and maximum individual dose) for jobs performed by outside organisations, it has to provide the outside workers with sufficient amount of individual protective equipment and to instruct them how to use it. Furthermore, the Ignalina NPP has:

- to record and to control the individual exposure results of outside workers;
- to control the radiological situation in workplaces and to provide recommendations how to perform the works safely;
- to record the results of workplace monitoring and individual monitoring and to provide the information to the outside workers and administration of outside organisation;
- if the outside worker may receive internal exposure, to perform the monitoring of committed dose (to estimate the internal dose by the whole body counter);
- to present the data about the outside workers exposure to the administration of outside organisation not rare than once per quarter and not later than 15 days after the outside worker has finished the work at Ignalina NPP.
- to record the individual dose of each outside worker in the Passbook of Outside Workers' Exposure.

The Ignalina NPP personnel also directly interacts with outside workers in following areas:

- in training and education in the field of radiation protection;
- in planning of works that are performed within the controlled area of Ignalina NPP;
- in evaluation of exposure results and planning of measures for exposure reduction.

It shall be recognised that proper training and education of outside workers is one of factors that allows to enhance the radiological safety culture at the Ignalina NPP. Training and education of outside workers in the field of radiation protection is organized according to requirements set out in [5]. The established frequency of training is 5 years. Training programmes prepared for training of Ignalina NPP workers are also applied for the training of outside workers. The programmes are approved by the Radiation Protection Centre. The duration of training is 30 hours. The contents of the training programme are presented in Table 1.

Additionally, according to requirements of [5], every year the outside workers are instructed by the personnel of Work Safety Department of Ignalina NPP in the field of radiation protection according to programme approved by the Ignalina NPP administration.

The Ignalina NPP provides the outside workers with the material needed for training in the field of radiation protection. Improving the level of qualification of outside workers in radiation protection area, the Ignalina NPP and the Radiation Protection Centre are actively participating in organizing of various training courses, issuing the radiation protection handbooks etc. Various projects are implemented in order to enhance the level of knowledge of outside workers in the field of radiation protection. Jointly with Swedish Nuclear Training and Education Centre (KSU), the Ignalina NPP and

the Radiation Protection Centre have prepared and adopted the educational book "Radiation Protection" that is also used for training of outside workers.

Table 1. Contents of the training programme

Item No.	Subject	Number of hours
	<i>Theoretical training</i>	
1	Basics of nuclear physics	2
2	Types of ionising radiation and measuring quantities	2
3	Biological impact of ionising radiation to humans	3
4	Sources of ionising radiation at Ignalina NPP and their characterisation	2
5	Methods for protection against ionising radiation	4
6	Basic principles of radiation protection. ALARA principle	4
7	Legal base of radiation protection	2
8	Organisational – technical measures ensuring radiation protection when conducting activities within the controlled area	3
9	Dosimetric control at Ignalina NPP	2
10	Decontamination, collection, storage, transport and disposal of radioactive waste	2
11	Duties and responsibilities of workers in case of radiological accident instructions on emergency preparedness at Ignalina NPP	2
	Total	28
	Practical training	
12	Methods for protection against ionising radiation	1
13	ALARA principle	1
	Total:	30
	Examination	2

The outside workers are directly involved in planning and evaluation of works that are performed within the controlled area of Ignalina NPP, in planning of doses and evaluation of exposure results, in planning of measures for exposure reduction. Involvement of outside workers in planning activities also allows to enhance the radiological safety culture.

The occupational exposure results can be used as indicator for evaluation of achieved level of radiological safety culture. The occupational exposure results of Ignalina NPP and outside workers during the year 1997-2001 are presented in Annex 1.

Conclusions

One of crucial factors that helps to enhance the radiological safety culture between the license holder, employer and outside workers is the clear allocation of tasks and distribution of responsibilities in the field of radiation protection.

The occupational exposure results can be used as indicator for evaluation of achieved level of radiological safety culture.

References

- [1] Lithuanian Hygiene Standard HN 73:2001 “Basic Standards of Radiation Protection” (2001).
- [2] Lithuanian Hygiene Standard HN 83:1998 “Radiation Protection and Safety of Outside Workers” (1998).
- [3] Government Resolution No. 653 of the Republic of Lithuania “On Regulations of Licensing the Practices Involving Sources of Ionising Radiation” (1999).
- [4] Order of the Minister of Health of the Republic of Lithuania No. 335 “On Procedure Submission about Sources of Ionising Radiation and Workers Exposure to the State Register of Sources of Ionising Radiation and Workers Exposure” (1999).
- [5] Order of the Minister of Health of the Republic of Lithuania No. 171 “On Procedure of Mandatory Training and Instructing for Persons, Responsible for Radiation Protection and Workers, whose Work Involves Sources of Ionising Radiation” (1999).

Annex I

Occupational exposure results of Ignalina NPP workers and outside workers (1997-2001)

	Ignalina NPP workers					Outside workers					Ignalina NPP workers and outside workers				
	1997	1998	1999	2000	2001	1997	1998	1999	2000	2001	1997	1998	1999	2000	2001
Number of workers	3232	3268	3315	3269	3187	986	956	830	575	1188	4218	4224	4145	3844	4375
Collective dose, man-mSv	11457	11481	10333	8516	5097	7046	3578	2458	2191	1180	18502	15059	12791	10707	6277
Average individual dose, mSv	3,55	3,51	3,12	2,61	1,60	7,15	3,74	2,96	3,81	0,99	4,39	3,56	3,09	2,79	1,43
Maximum individual Dose, mSv	42,47	37,51	31,89	24,02	19,30	44,85	36,80	33,30	19,66	18,41	44,85	37,51	33,30	24,02	19,30
Dose range, mSv	Number of workers														
0.0-0.5	1482	1355	1716	2041	2165	485	465	482	345	1007	1967	1820	2198	2386	3172
0.5-1.0	513	526	318	209	225	100	98	78	33	33	613	624	396	242	258
1.0-5.0	657	717	617	478	439	144	188	116	44	69	801	905	733	522	508
5.0-10.0	201	293	326	197	200	34	67	64	40	37	2351	360	390	237	237
10.0-20.0	217	254	237	296	158	64	92	68	113	42	281	346	305	409	200
20.0-30.0	85	85	91	48	-	55	31	12	-	-	140	116	103	48	-
30.0-40.0	72	38	10	-	-	59	15	10	-	-	131	53	20	-	-
40.0-45.0	5	-	-	-	-	45	-	-	-	-	50	-	-	-	-
>45.0	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

ABOUT THE INTEREST OF AN AGREEMENT FOR A EUROPEAN PASSPORT FOR WORKERS IN EUROPEAN NUCLEAR FUEL CYCLE INSTALLATIONS

M. Gonin, C. Bailloeuil, A. Pétrequin, M. Teissier
EDF-Gaz, France

Abstract

The European directive 96/29 applicable to EEC member states is, or will be transposed in each country according to national conditions which could prove to be more restrictive than the demands of the directive.

Additionally, specific organisations in each country, reinforce the disparities where radioprotection, and medical and dosimetric follow-up are concerned.

In April 2001, initiated by a group of French company medical officers (EDF CEA COGEMA), a meeting with company medical officers of the EEC: Great Britain, Italy, Spain, Belgium, Germany, was organised in order to bring to attention the issue of the movement of contract workers in European nuclear power plants.

The findings were that, although the rules, procedures and organisations might be very different, the objectives of medical and dosimetric follow-up remain the same.

The group proposes the creation of a “European Worker Passport”, with knowledge and agreement of all.

This passport, property of workers, could contained (apart from indispensable official data on the employee, his employer, the company medical officer and the radiation monitoring service):

- dosimetric data;
- medical fitness data;
- level of training;
- establishment in return of a radiation pass book with an ongoing account of exposure received by the worker on the nuclear power plant.

The European directive 96/29 applicable to EEC member states is, or will be transposed in each country according to national conditions which could prove to be more restrictive than the demands of the directive.

Additionally, specific organisations in each country, reinforce the disparities where radioprotection, and medical and dosimetric follow-up are concerned.

In April 2001, on the initiative of a group of French company medical officers (EDF CEA COGEMA), a meeting with company medical officers from various member states of the EEC: Great Britain, Italy, Spain, Belgium, Germany, was organised in order to bring to attention the issue of the movement of contract workers in European nuclear power plants.

The findings were that, although the rules, procedures and organisations might be very different, the objectives of medical and dosimetric follow-up remain the same.

In summary:

- For the objective of dosimetric follow-up:
 - a regulatory individual dose limit of 20 mSv per year;
 - a career limit for certain countries, e.g. Germany, 400 mSv;
 - monitoring of external exposure by external dose measurement equipment (TLD, film badges...) considered as the legal dose recording, and by electronic dosimeter;
 - monitoring of internal exposure by whole-body counts and radiotoxicological measurement examinations using the recommendations of the ICRP to calculate the level of intake, under the responsibility of the radioprotectionist or health physics services or of the health occupational service according to country;
 - dosimetric data to be recorded either in medical files, or in a data base under the responsibility of specific authorities.
- For medical follow-up:
 - a medical certificate, for exposure to ionising radiation, issued by agreed or appointed doctors for a period of six months to a year; and
 - of professional qualification for exposure to tedious/tiring work;
 - establishment of a detailed job description and obligatory setting up of post-exposure and post-occupational follow-up, according to country;
 - obligation of employees to be trained and informed about the risks, as well as the basic principals of radioprotection.

There already exists in each country, a personal entry passbook for each employee. This entry passbook contains data relating to the following:

- official data concerning the worker;
- data regarding the employer;
- data from the company medical officer;
- level of professional training followed;
- professional qualifications;
- medical fitness for exposure to ionising rays, and even other risks;
- monthly and cumulative dosimetric data;
- dates and also results of examinations of internal dosimetry.

An additional document exists, edited under the aegis of the European communities by the ISPRA research centre in Italy.

It is also useful to note the case, peculiar to France, on the confidentiality of dosimetric data which cannot be communicated to the employer, as well as the responsibility of the company medical officer where internal dosimetry is concerned.

On the subject of the movement of contract workers within the European arena, it is necessary that medical and dosimetric follow-up should be more efficient and less redundant, and the best adapted to specific work situations in each nuclear power plant.

The objective of transferring data and keeping track of exposure to risks of whatever type, cannot be ignored.

A system for data transmission in both directions must be put into place.

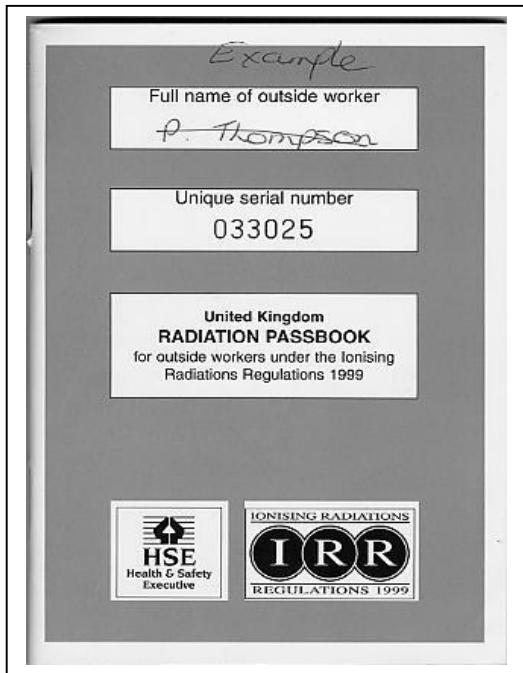
This system must be compatible with all member states.

In actuality, the question arises, of the validity of the information concerning each employee on different complementary aspects: authorization of access to controlled zones taking into account training, medical fitness, authorization to work under certain risks bound by specific rules (tedious work, heat, carcinogenic substances).

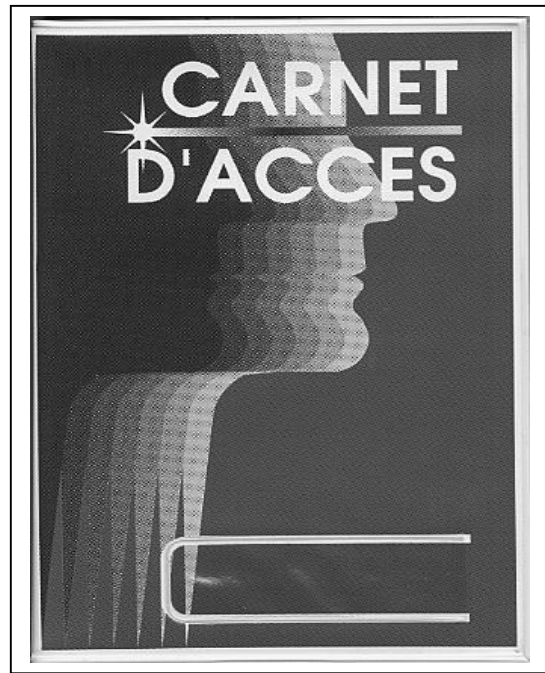
In return, the employee must be furnished with information regarding both the result of internal and external dosimetric follow-up carried out by the NPP, and the state of exposure to other risks.

The group proposes the creation of a “European Worker Passport”, property of the worker, containing apart from indispensable official data on the employee, his employer, the company medical officer and the radiation monitoring service, dosimetric data, medical fitness data, the level of training... as well as the establishment in return, of a radiation pass book with an ongoing account of exposure received by the worker on the nuclear power plant.

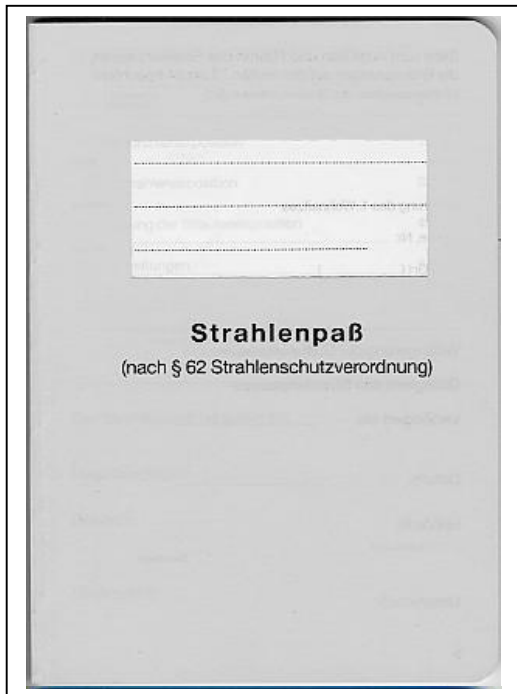
It must be independent of all national data concerning the occupational health of workers. The objective of the “European Worker Passport” is to facilitate movement of contract workers within the European area, to reinforce their dosimetric and health medical following in conformity with the law of each country.



Great Britain



France



Germany



UE-ISPRA (Joint Research Center)

A SWEDISH DOSE PASSPORT – CONTRACTORS POINT OF VIEW

M. Andersson, A. Holmqvist, J. Möller
Westinghouse Atom AB, Sweden

Background

Westinghouse Atom is situated in Västerås approximately 100 km west from Stockholm. The company is owned by BNFL. The two largest divisions are the Nuclear Fuel Operations and The Global Reactor Services division.

The Nuclear fuel operations manufacture fuel for BWR and PWR reactors. The raw material used is Uranium hexafluoride, which is converted to Uranium dioxide powder through the wet AUC-process. The concession is 600 tonnes of UO_2 per year. Last year the production was approximately 900 fuel elements.

There is also a control rod production line within the fuel factory. Last year the production of control rods was approximately 160.

The Global Reactor Services Division performs tests on different types of equipments used in nuclear power plants. In addition there is also a well-established service structure that provides a wide range of field services, for instance sipping of fuel elements.

The total amount of people working in Västerås is currently around 800. The majority of those, work at the fuel factory.

Westinghouse Atom
Nuclear fuel operations



Westinghouse Atom
Global reactor services



Our personnel working within reactor services visit nuclear power stations all around the world and hence, they are strongly dependent on a properly working dose registering procedure. In Sweden we have a system commonly owned by the Swedish nuclear installations. The system is called CDIS and is administered by a Swedish IT-contractor. According to Swedish regulation all the doses for workers have to be fed into the system on a monthly basis.

Purpose

The purpose of this paper is to describe the somewhat awkward situation for our employees when working as external personnel on German nuclear installations. Our Swedish personnel are currently using German dose passports. Since Sweden joined the European Union in 1995 this is in contradiction to the EU-directives. Hence, Westinghouse Atom has applied for a license for the use of Swedish dose passports in Germany. The amount of people performing service jobs in Germany is approximately 80 persons.

Description of the procedure for German dose passports

Today, when our employees are travelling to Germany for field service missions they are using a German dose passport, which is administered and issued by Westinghouse Reactor in Mannheim.

If a new German dose passport is to be issued Westinghouse Atom has to send a dose excerpt from the Swedish registry, a witnessed copy of the persons passport and a signed German dose passport to Westinghouse Reactor. This procedure takes approximately 1 month for a new dose passport.

Westinghouse Reactor sends the dose passport to the respective nuclear power plant along with the film dosimeter. When our person has completed his work and leaves the nuclear power plant the dose passport and the dosimeter are returned to Westinghouse Reactor in Mannheim. After 3 months health physics office at Westinghouse Atom receive a letter stating the dose received by the employee during his period of work at the power station. If our employee is performing multiple works on several power stations he takes the German dose passport along with him until he has finished his last job. The health physics office on the last installation returns the dose passport to Westinghouse Reactor.

As a contractor it is important to apply for a license to use Swedish dose passports in Germany because:

1. we want to be responsible for our personnel that we send to Germany;
2. the results from the dose evaluation will be obtained faster;
3. cost savings are substantial;
4. the administration would be minimised.

Description of the Swedish dose passport

The competent authority, Swedish radiation protection institute (SSI), issues dose passports in Sweden. The only data required is an excerpt from the Swedish dose registry, CDIS, stating the effective dose over the past five years, the annual effective dose and the life time effective dose. The SSI then needs a maximum of 3 days to process the request.

The Swedish dose passport is somewhat different from the German dose passport. The Swedish dose passport is only valid for a period of 6 months. The German dose passport is valid for a lifetime or until the dose passport is filled with records. Unlike the German dose passport the Swedish competent authority issues the Swedish passport. The Swedish authority does not require the result from the radiological health examination. They consider the result as personal and have therefore chosen not to state the result in the dose passport. Consequently, our external personnel are obliged to bring their medical examination record and the Swedish dose passport when they go on their service missions to Germany.

The legal basis for the Swedish dose passport is found in the ordinance SSI FS 1996:3, published by the SSI.

Description of the licensing procedure to use Swedish dose passports in Germany

The EU-directives 90/641/Euratom and 80/836/Euratom provides the basis for the use of dose passports for external personnel working at nuclear facilities in the European Community.

The Swedish application was outlined in accordance with the requirements from the German competent authority Bayerisches Landesamt für Umweltschutz (LfU). The following data had to be submitted:

1. Application according to §20 Radiation Protection Ordinance.
2. Declaration form for the person responsible for radiation protection.
3. Letter from the Swedish radiation protection institute stating that Westinghouse Atom has a dosimetry service compliant to the Swedish regulations.
4. Excerpts from the Swedish criminal record for the president of Westinghouse Atom and the person responsible for radiation protection.
5. Documents confirming the educational level of the person responsible for radiation protection.
6. Document stating the legal relationship between Westinghouse Atom in Västerås and Westinghouse Reaktor in Mannheim.

German legal demands for us as a contractor using Swedish dose passports

The legal demands on Westinghouse Atom as a contractor on German nuclear installations are based on the German Radiation Protection Ordinance.

In our case we sent the application to LfU in München. They processed the application and agreed on a license for a 5 years period. However, the license is coupled to several demands before the license can enter into force in Germany.

1. All of the German nuclear installations that we visit must sign an agreement with us on the health physics issues. The holder of the license must prove to LfU that an agreement, regarding administrative and organisational issues related to health physics, has been made between Westinghouse Atom and the relevant German nuclear installations.
2. The holder of the license has to present a radiation protection instruction regarding the internal routines relevant for health physics.
3. All of our external personnel have to be listed. The holder of the license is responsible for ensuring that the listed workers receive proper information about the conditions on the relevant nuclear installation.
4. The holder of the license must, without delay, inform the person responsible for radiation protection of the nuclear installation if the activity- or dose limits are exceeded.
5. The holder of the license shall:
 - measure the dose with a TL-dosimeter qualified by the Swedish competent authority, SSI;
 - make sure that the listed workers carry the dosimeters provided by the installation;
 - let the workers go through whole-body measurement at LfUs qualified measurement station for radio toxicology.
6. The holder of the license has to make sure that the external personnel follow the instruction given by the health physics office on site.
7. The holder of the licence must update the Swedish dose registry as well as the dose passports.
8. The results from the monthly dose evaluation of the listed workers have to be forwarded to LfU every month.
9. The list of workers has to be sent to LfU once every third month or when there are changes done to the list.
10. The Swedish dose passports have to be registered at the LfU.
11. When a listed worker stops working at German nuclear installations, the dose passport shall be left with him.

The agreement between Westinghouse Atom and the nuclear installations has not yet been reached. Therefore, we are still working with German dose passports for our external personnel.

Summary

Westinghouse Atom has applied for a license to use Swedish dose passports in Germany, however there are still some issues to deal with before the license can enter into force.

The procedure for making Swedish dose passports a reality in Germany has been a lengthy process, but with the good and continued help from the German competent authority, LfU, the process will hopefully be concluded in the near future.

Finally, I ask myself:

Could a European – or even global dose passport simplify or even eradicate these situations from arising in the future for all of us?

THE BNFL LEGAL ELECTRONIC DOSIMETRY SERVICE

A. Weeks*, M. Pottinger*, P. Clarke#

*BNFL, Research and Technology, United Kingdom.

#BNFL Magnox Generation, United Kingdom

Introduction

BNFL Magnox Generation started a three year scheme in April 1996 to introduce Siemens Electronic Personal Dosimetry (EPD) systems into its reactor sites as part of an initiative to improve the control of doses and the accuracy of dose statistics and to record personal legal dose.

Concurrent with the installation of the EPD systems a successful application was made to the United Kingdom Health and Safety Executive (HSE) for approval of the BNFL dosimetry service to use the Siemens EPD Mk 1.2 for recording legal doses. This paper discusses the experiences of the BNFL dosimetry service in operating the approved dosimetry service since it's approval by the HSE in January 2000.

Outline of the approved dosimetry service

Under the auspices of regulation 35 of the UK Ionising Radiations Regulations 1999 (IRRs) the Health and Safety Executive have issued a series of documents, "Requirements for the Approval of Dosimetry Services (RADS), consisting of, Part 1: External Radiations, Part 2: Internal Radiations and Part 3: Co-ordination and Record Keeping. This paper relates specifically to the approval of the Service to assess external radiations using EPD under RADS Part 1 External Radiations.

A dosimetry service must show that it can meet the criteria specified in the RADS in order to obtain approval. Applications for approval are assessed on the basis of a Statement of Service provided by the applicant and information gained by the HSE on the Service's organisation, resources, personnel and methods and from reports of performance tests.

The RADS have been written around the concept of the passive dosimeter, but within the guidance the HSE have made provision for the introduction of new technologies such as EPD. Prior to submitting an application for approval the Service and the HSE discussed in detail the format of the Statement of Service and its ability to comply with the RADS. It was concluded that the requirements detailed in the RADS could be followed for active or passive dosimeters.

The Service who are located within BNFL Research and Technology at Berkeley Centre, holds a central approval for the assessment of EPD data from its Clients, who are the BNFL Magnox Generation and British Energy reactor sites. The Client must obtain approval from the Service and the HSE for use of the EPD as the legal dosimeter. This approval is given subject to a satisfactory quality audit of their systems and procedures in order to satisfy the Service and the HSE that they are compliant with the appropriate sections of the Service's Statement of Service. All audit results will be

forwarded to the HSE. On completion of a successful audit of a Client the Service will formally notify the HSE in writing. The HSE will include the name of the approved Client on the Services certificate of approval.

Every calendar month, all dose results, dose estimates, and anomalous results are sent, via electronic transfer, to the Service from its Clients. The Service assesses doses on a Client by Client basis by summation of individual working session results, taking into account any dose estimates. Each working session result will be verified and any anomalous results will be investigated by the Service. All investigations will involve the Client before ratification of results by the Service.

The Service forwards monthly electronic summaries of the dose results to the BNFL Central Dose Records Service (CDRS) where assessed doses are collated for entry on to the Dose Records. CDRS is the BNFL Approved Dosimetry Service for Co-ordination and Record Keeping for reactor sites.

EPD systems

The Siemens EPD was developed in the UK in collaboration with the National Radiological Protection Board. (NRPB). The EPD has been designed to measure photons in the energy range 20 keV to 6 MeV, and beta radiation in the energy range 250 keV to 1.5 MeV. It utilises 3 silicon diode PIN detectors, and incorporates sophisticated micro-circuitry powered by a bespoke high-energy battery designed to last at least a year under normal operational use. The EPD stores raw count data from the three detectors in four channels: hard gamma (HG), soft gamma (SG), full beta counts (FB) and beta compensated counts (BC). From this data, the EPD evaluates the personal dose equivalents $H_p(10)$ and $H_p(0.07)$. This data is stored to secure memory every 15 minutes to minimise data loss on battery or other failure.

The Dose Control Software, DCS-3 manages the EPD system on Client sites it stores information on Oracle relational database tables. The EPD wearer and the Task ID are the main set up parameters for dose management with the EPD wearer being the key set up for legal dose measurement. The DCS software requires all legal information such as a valid medical date to be entered before a wearer can issue an EPD. Additional compliance for parameters such as training validity can be introduced by the site. Personal details for individuals are mainly standard entries such as name, date of birth and National Insurance number. Dose Credit ID's are selected for each individual for dose control and regulatory compliance. Department ID can be set up as a four digit code to ease the retrieval of information using relational database query tools.

An individual will issue an EPD by selecting an EPD from a rack and inserting it into a slot in an Access Control Work Station (ACW). The screen instructions will request the individuals Personal Identification Number (PIN) and a Task ID for the proposed work. The Task ID will define the dose and dose rate alarm levels programmed into the EPD on issue. The DCS-3 provides powerful control as it will not issue an EPD to a person unless all compliances such as Medical and Dose Credit are satisfactory.

Quality assurance

The Service's approval is based on its ability to produce an auditable quality system for ensuring that individual doses recorded in the workplace can without exception be entered onto the correct dose record, having first been independently verified. The Service's Statement of Service

states that the issue and control of EPD's at Client sites is covered by the Clients own quality assurance procedures. All Clients prepare an Operating Manual and suit of work instructions which meet their own QA requirements. The contents of the Operating Manual are set out in the Services Statement of Service. The Service will audit Clients against the Operating Manual before applying to the HSE for the Client site to be added to the Services certificate of approval.

Prospective Clients of the Service will submit the documentation discussed above to the Service who will initially carry out an audit of the documentation before visiting the Clients site to audit the EPD systems. Any short comings or non-conformances within the documentation will be raised as a corrective action and discussed with the Client during the site visit. During the site visit the Service will interview key personnel involved with the day to day operation of the EPD systems including staff responsible for operations during silent hours and staff providing IT services. Checks will be made on training records for interviewed staff and the general training arrangements for EPD users. The Service will also look at site protocols for recovery of the computer systems in the event a partial or complete failure and procedures for making dose estimates. Any non-conformances or short comings in the site audit will also be raised as corrective actions.

At the audit closing meeting the Service will present the Client with it's finding in the form of corrective actions and observations. The Client will be asked to sign on to the corrective actions and propose a solution and a completion date for each action. The Service will then formally issue an audit report.

When the Client site has made the appropriate modifications to it's systems to enable it to discharge any corrective actions all relevant information and documentation will be sent to the Service. The Services will review the documentation and evidence supplied by the Client and if satisfied will sign off the corrective actions. An application will then be made in writing to the Health and Safety Executive for the client site to be added to the Services Certificate of Approval. Before issuing the certificate of approval an HSE inspector will normally visit the Client site.

The most commonly occurring corrective actions from the 12 audits carried out by the Service to date relate to the recording of training on specific work instructions, which is fundamental to the success of using electronic dosimeters for legal dosimetry. Many sites have now made improvements in recording work instruction training in dosimetry and other areas as part of their site licence requirements for all staff to be suitably qualified and experienced persons (SQEP).

Another common problem has been the development of robust systems for dealing with a computer system failure either partial or catastrophic. Generic service contracts covering the rebuilding or replacement of systems within given time scales are held by Clients with Siemens in the event of failure. Several Clients had not developed sufficient systems to cope with getting people in and out of the controlled area in the event of a computer systems failure. Procedures had to be developed and staff trained to allow a manual entry system to be before the EPD could be used for legal dose assessment.

Data verification and diagnostics

Data received by the Service from Client sites will undergo several tests and selective verification routine before being assessed. The diagnostics tests allow the assessor to select and evaluate the data of specific interest. The number of working session doses a Magnox site can generate in a month will range from about 3 000 to 25 000 The larger British Energy sites can generate up to 50 000 visits a month during outages. As most working sessions generate doses of less than 5 μ Sv it is

essential for the efficiency of the service to be able to automatically select results of specific interest for further investigation.

An important diagnostic tool developed by the Service is the ability to confirm spurious doses by obtaining detector channel ratios from the raw data. The ratio technique can be used for general guidance when assessing working session data. The ratios will enable the assessor to flag possible problems and if necessary instigate an investigation with the Client. The Client can confirm, for example, the characteristics of the radiation field the individual was working in at the time the dose was received and other relevant information.

The EPD stores raw count data in four channels: hard gamma (HG), soft gamma (SG), full beta counts (FB) and beta compensated counts (BC). From this data, the EPD evaluates the personal dose equivalents $H_p(10)$ and $H_p(0.07)$. Each radiation environment has a fingerprint which is characterised by the following ratios: HG/SG, SG/FB and FB/BC. Figure 1 graphically illustrates the HG/SG ratio data for operational Magnox, Advanced Gas Cooled Reactor (AGR) and Pressurised Water Reactor (PWR) plants, also a decommissioned Magnox plant all using the Mk 1.2 EPD. Data is also shown for Hinkley Point B AGR which uses the Mk 2 EPD. Figure 2 shows a bar chart of HG/SG for most of the UK reactor sites.

From Figure 1, it can be seen that the peaks of the probability curves for the AGR and PWR reactors are sharper than those of the operational and decommissioning Magnox reactors. This is due to the more diverse nature of photon radiation energies that make up the ambient radiation fields on the Magnox plants, due in the main to less efficient shielding when compared to an AGR. The shape of the AGR and PWR curves is due to the mainly outage related ^{60}Co dominated radiation fields that make up the majority of personal doses on these sites.

Figure 2 shows that the magnitude of the ratios for the Mk 1.2 EPD varies from 2.67 at Dungeness B to 3.83 at Chapel Cross. The range of the ratios is dominated by the ambient photon energy. Chapel Cross is the oldest reactor (Magnox) in the group and as such the shielding associated with the reactor is not as efficient as the more modern AGR reactors. This results in a high energy ambient photon (6.1 MeV) radiation field from the exposed gas coolant ducts which contributes significantly to personal doses and also increases the value of the HG/SG ratio. However Hutton B a more modern AGR power station which has excellent shielding, also has a fairly high HG/SG ratio because most of the dose associated with the plant is acquired during vessel entries (^{60}Co 1.3 MeV). Dungeness B (AGR) also has excellent shielding and hence well moderated ambient photon energies, which combined with their 2001 outage not involving vessel entries gave much lower HG/SG ratios.

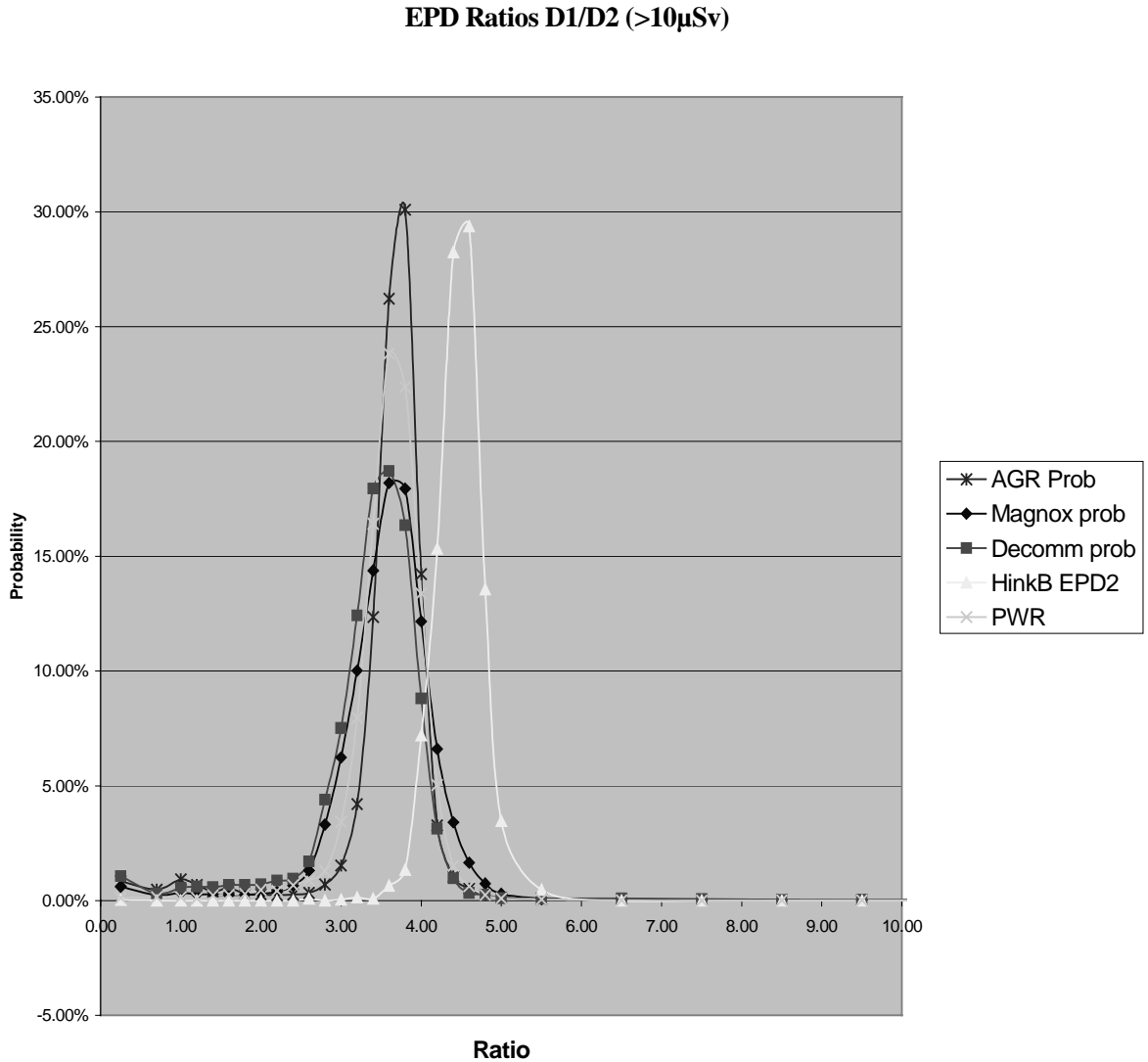
Management of dose estimates

The IRRs regulation 22 specifies the requirements that must be met for estimating a dose when information is lost for whatever reason. The Service and Clients must ensure that robust management arrangements are in place to meet all the IRR requirements. As part of the management of doses on site amendments may be required. These normally fall into three categories.

Manual amendments

Amendments made due to EPD failure in the controlled area, most commonly due to physical damage which renders the dosimeter unreadable or to a lesser extent battery failure.

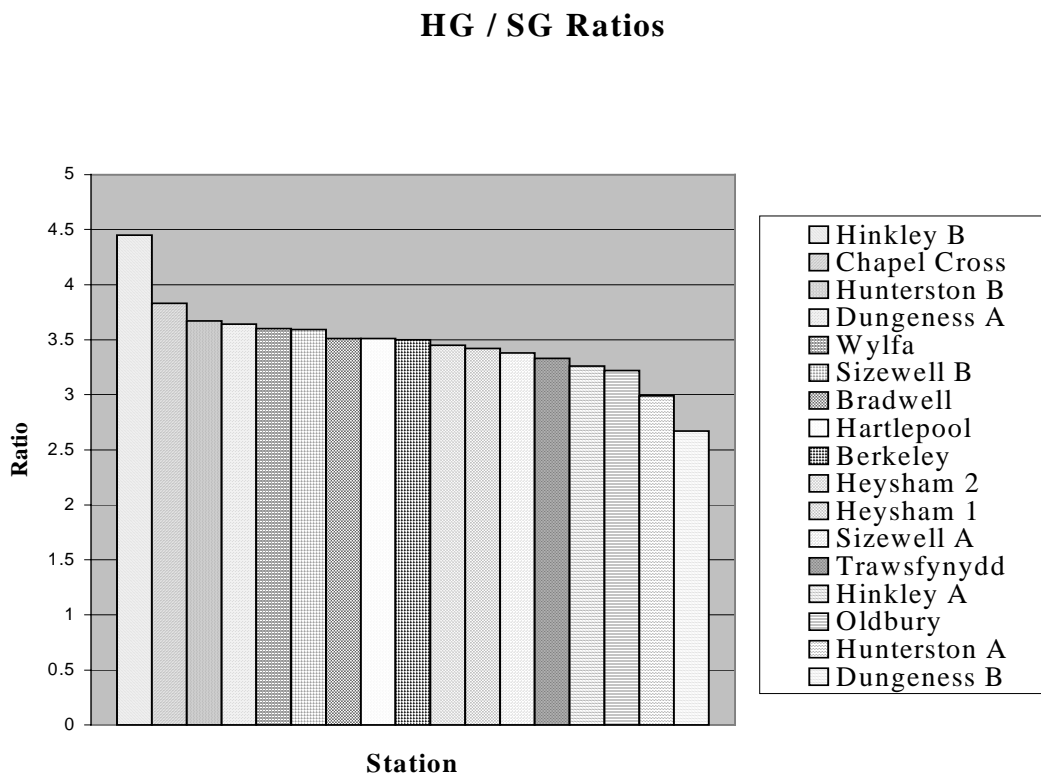
Figure 1. HG/SG by reactor type (Note: data for Hinkley B is for the Mk 2 EPD)



Site amendments.

Amendments made by the site to spurious doses caused for example by radio frequency interference (RFI) from electrical appliances and security tags, electrostatic conduction or light leakage through the beta window. In these cases the dosimeter will be readable but will have an enhanced dose reading.

Figure 2. Ratio of HG/SG by reactor site (Note: data for Hinkley B is for the Mk 2 EPD)



Dosimetry service amendments.

Anomalous doses identified by the Services diagnostics software. These dosimeters will have been read in the normal way but the spurious nature of the dose may not have been picked up by the wearer or the site Health Physics staff. The causes of these spurious doses is usually the same as discussed in 2 above.

In cases 1 and 2 above the site will have performed a dose investigations prior to transmitting the dose to the Service. These investigations are often carried out at the time of, or shortly after the incident. If the amendment required is manual the individual will not be able to re-enter the controlled area due to an incomplete working session (i.e. unable to read the dosimeter on exit) this will require a manual amendment to complete the Working Session. In the case of a spurious dose due for example to RFI there is normally a dose or dose rate alarm associated with the incident which the wearer should report to the Health Physics staff. However this is not always the case as the spurious dose may occur in a high dose rate area where dose alarms may be expected. When the ADS finds doses it thinks are anomalous it informs the Client site giving full details of all information making up the working session. Due to the nature of the diagnostic software the Service can in some cases give guidance on what the true dose may be, if for example a spurious dose is incremented with a real dose. The Client site will then perform an investigation, if they are satisfied that the dose was anomalous they will notify the Service who will amend the record in question before entry onto the legal dose record. Any records amended by the Service are electronically transmitted to the Client site to enable an update of their database. The records are flagged as being amended by the Service.

The Service receives an electronic version of dose investigations performed for both manual and site amendments. For manual amendments the Service has no other information other than the investigation report and the doses. However for site amendments the Service is able to run the original EPD data through the verification software, this allows the Service, to independently verify the sites amended data.

Detailed below in Table 1. is information on the number of amendments made during 2001 for various sites of differing reactor types also shown in the table are the numbers of working session doses above 10 μ Sv for 2001. Also data for the Mk2 EPD for Hinkley Point B Power Station, which clearly shows an improvement in reliability and sensitivity to radio frequency interference.

Table 1. Dose amendments for 2001

Station	Reactor type	Number of visits in 2001	No. manual amendments	No. site amendments	No. ADS amendments	No. of visits above 10 μ Sv (% of Total)
Trawsfynydd	Decommissioned Magnox	51261	161	1038	31	3259 (6.36 %)
Dungeness A	Magnox Steel Pressure Vessel	108106	237	1522	316	4594 (4.25%)
Dungeness B	Advanced Gas Cooled (AGR)	126695	262	2360	11	136 (0.12%)
Oldbury	Magnox Concrete Pressure vessel	75424	154	646	78	1177 (1.56%)
Sizewell B	Pressurised Water Reactor (PWR)	55333	47	2035	12	2484 (4.49%)
Hinkley Point B	AGR Mk 2 EPD	91353	47	2	0	3360 (3.68%)

Summary of systems performance

The BNFL Approved Dosimetry Services at Berkeley was issued with it's first certificate of approval with Oldbury Power Station as its first client in January 2000. Oldbury has been using the approval since 1 June 2000 following the cessation of film badge issue on 31 May 2000. Since that time the HSE have given approval to 4 more sites, Wylfa (Magnox) Dungeness B (AGR), Hunterston B (AGR) and Sizewell B (PWR). The HSE is currently considering the applications of 7 more Clients of the ADS who should obtain formal approval over the next few months.

The experience of the Service to date has confirmed that metrologically characteristics the Siemens Mk 1.2 EPD performs well in all radiological environments encountered on operating and decommissioning nuclear power plant. It's low threshold of detection has enabled Client sites to accurately control doses to much lower levels. The use of EPD has also removed the sometimes large statistical uncertainties in calculating site and group collective doses associated with passive dosimeters.

In terms of operating costs the Service and Client sites now require fewer personnel to operate their services, this has proved to be expedient with the downsizing of the nuclear industry in the UK over the last few years. The Mk 1.2 EPD has not been as reliable as first anticipated due to its sensitivity to radio frequency interference and poor battery performance which has proved to be administratively expensive. The return rate for repair of damaged EPD's in the BNFL Magnox population of 2 800 dosimeters is currently running at about 20% per annum, which compares to 2%

for the Hinkley Point B population (500 EPD's) of Mk 2 EPD's. However the Access Control Work Stations (ACW) and computer systems have proved to be very reliable and robust.

The future

Before the end of this year it is anticipated that the Service will have 16 Client sites (all reactor sites) approved for the use of the Mk 1.2 EPD as a legal dosimeter. The Service will also make an application during 2002 to the UK Health and Safety Executive for an approval for the Mk 2 EPD using Hinkley Point B as it's first Client site.

During the financial year 2003/2004 BNFL will consider a project to upgrade it's current stocks of Mk 1.2 EPD and associated equipment to the Mk 2 EPD against commercial benefits and other options. Concurrent with this the Service will seek to obtain appropriate approvals from the HSE for it's clients moving to the Mk2 EPD.

CDRL – COMPANY DOSE RESTRICTION LEVEL

S. Morris

HSED, British Energy Generation Ltd, United Kingdom

Introduction

For a number of year's dose constraints and controls have been used as effective measures in aiding restricting exposure to ionising radiation. Predecessor companies to British Energy Generation (BEG) originally established the Company Dose Restriction Level (CDRL) as a consequence of the revision of risk estimates, then with the revised Ionising Radiations Regulations 1999 (IRR99) [1] the CDRL for BEG was also revised. The background, influences and consequences of CDRL appliance in a commercial organisation in calendar year 2000/1 are presented below.

Background

The British Energy Group is an energy company, which operates 15 nuclear units in GB supplying about 22% of the electricity market [2]. One BEG site, Sizewell B, has one Pressurised Water Reactor (PWR) whilst the remaining seven sites each have twin Advanced Gas cooled Reactors (AGR).

The annual collective dose trend for the previous 6 years for each site is provided in fig.1. Evidently three BEG locations predominate Company dose, Sizewell B, Hinkley Point B and Hunterston B. Hinkley and Hunterston are sites where entry to the reactor vessel is routinely performed during their triennial outage periods; other AGR locations rarely undertake vessel entry. Typically 90% of a BEG sites' annual dose is accrued during outage periods.

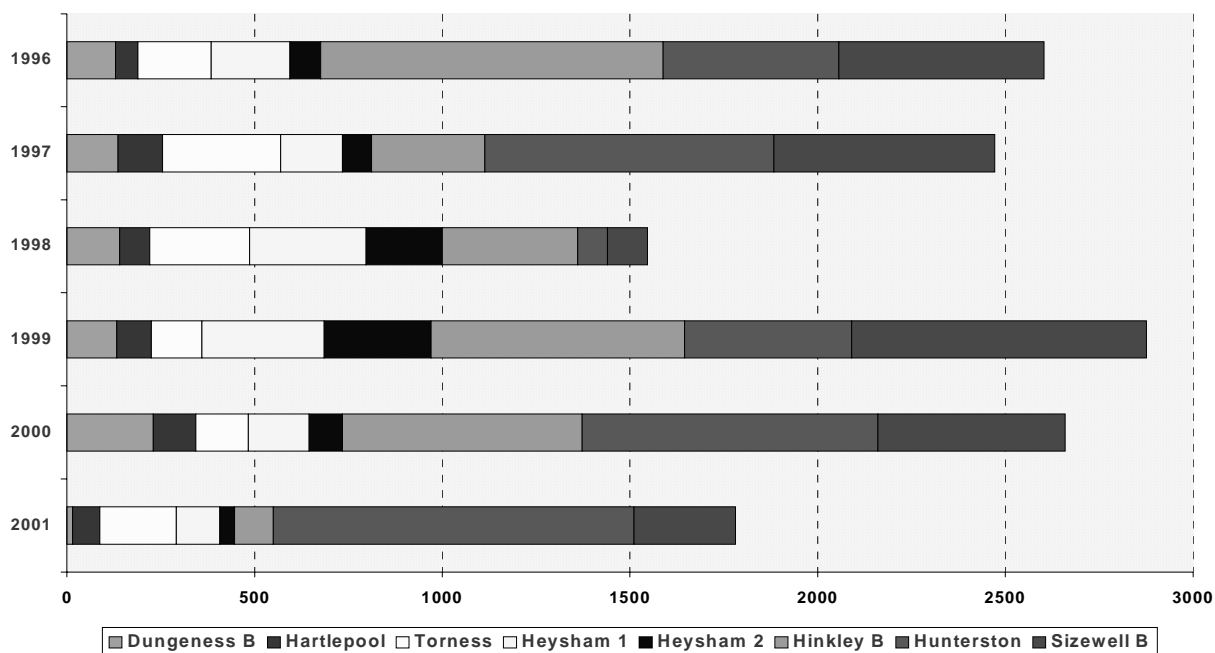
Internal influences for CDRL development

Predecessor companies to BEG formulated a CDRL to restrict radiation doses to employees and contractors to 15 mSv in any calendar year. This CDRL was imposed irrespective of where exposure to radiation was received, so was not source-related. The CDRL should not be misunderstood as an exposure limit since the restriction level can be exceeded, but only under very particular circumstances. It is important to note that by ensuring individuals did not exceed the CDRL there would not be a need to perform the annual or 5-year investigations required by statutory provisions. There has been only one case of an individual exceeding the 15 mSv CDRL since its introduction in 1991 up to 1999.

The BEG Company Radiation Protection Adviser (CRPA) decided to use the opportunity of the introduction of the IRR99 [1] to completely revise the Radiological Safety Rules and Instructions. The CRPA also considered the implications for changes to the CDRL based upon the reduced

statutory dose limits, its prior success in dose management and upon previous and future maximum individual dose across BEG sites (where Sizewell B was considered to be the restraining case). Apart from a reduction of the CDRL, due consideration was given to the constraint being source-related such that the CDRL was made applicable to doses received whilst specifically working on BEG sites. This provides a non BEG individuals' employer the flexibility to establish their own dose constraint and dose limit philosophy.

Figure 1. BEG annual collective dose by site in man mSv



BEG Company Dose Restriction Level (Table 1)

The CDRL is the annual maximum dose that can be planned for any employee, or contractors' employee to receive from work on BEG locations. In exceptional circumstances, (and with the agreement of the contractor) the BEG Company Executive Director of Health & Safety (HSED) may authorise (or approve) the exposure of personnel to doses which are greater than the CDRL provided doses are maintained as low as reasonably practicable (and remain within statutory dose limits). BEG adopted under the IRR99 [1] the statutory effective dose limit of 20 mSv per year.

Table 1. Company dose restriction level

Category	Restriction level
Employees and contractors aged 18 years or over (not being a trainee or other person).	10 mSv effective dose per calendar year.
Any female employee who has informed her employer that she is pregnant.	1 mSv equivalent dose to the surface of the abdomen for the remainder of the pregnancy.

Dose constraints and pre-work ALARP assessments (Table 2)

Within BEG a dose constraint is defined as the dose that is anticipated for a particular task at the planning stage. Dose constraints can be expressed as the dose to an individual team member, or the collective dose to all persons involved in the task. The Rules require dose constraints to be specified in formal ALARP reports that have to be prepared in advance of major projects involving exposure to radiation.

Table 2. Pre-work ALARP assessment requirements

Action Level	Action	Responsibility
All work in controlled areas	Carry out an ALARP assessment as part of the work planning process. Where it is planned that a person will receive a dose of more than 0.5 mSv in a month from the task. The radiation protection supervisor must be notified. A health physicist may provide advice on dose reduction practices.	S.Q.E.P. Radiation Protection Supervisor
Predicted dose for work greater than 3-mSv individual or 10-man mSv collective.	An ALARP report must be prepared at the planning stage to ensure that doses are kept ALARP. The report must contain a dose constraint for the work.	Health Physicist
Predicted dose for work greater than 6 mSv individual or 100-man mSv collective	The ALARP report required by the section above must be sent to the Director of HSED for review prior to work commencing.	Station Health Physicist
Predicted dose for work greater than 10 mSv individual.	No work should be planned to exceed any CDRL. Only where a justification and ALARP case can be made in exceptional circumstances will sanction to exceed a CDRL be given.	HSED Director

Pre-work ALARP assessments must estimate radiation doses to groups of workers for activities performed across **all** BEG sites. Clearly, where contract staff is employed, it is prudent to co-operate with the employer, share dose constraint policy and predicted dose levels. In some cases, because of the special nature of the work or the skills required by key specialist staff, including foreign nationals, exposures may approach a CDRL. Dose management and budgeting is exercised, with a clear profile of activities and dose uptake. Dose estimates should be maintained during the course of the work by using direct reading dosimeters to give real time dose information and allow for adjustments to be made to the work procedures. Work must incorporate a contingency factor to ensure that a CDRL is not approached or planned to be exceeded. In exceptional circumstances the Director HSED may authorise exposure of personnel to doses above the CDRL, provided doses are maintained ALARP and below statutory limits. Before sanction is provided a review is undertaken where justification, ALARP, future work and special circumstances are considered. This information is required to support the subsequent mandatory requirement for an investigation. Pre-work ALARP assessments contain as a minimum the following elements:

- an overview of the work and why it has to be done;
- an assessment of alternative methods of working;

- a prediction of the dose and dose rates associated with the work;
- lessons learnt from previous work of a similar nature; and
- actions to be taken to keep dose ALARP.

From policy to practice Hunterston and Hinkley routine outages

BEG formed an approved strategy with a single contractor to tackle the inspection and repair activities of in reactor vessel boilers at both Hunterston and Hinkley during 1999 and 2000. This comprises the routine outages for all four reactors including both planned inspection and repair work (where the dose can be predetermined) and some emergent work (where the degree of work and dose is indeterminate). The basic approach to planning, training, deploying and managing a large and diverse team across 2 sites was considered sound. Decisions on the extent of emergent work is dictated by factors such as defect severity, location, repair techniques, programme, manpower, safety case implication and commercial risk.

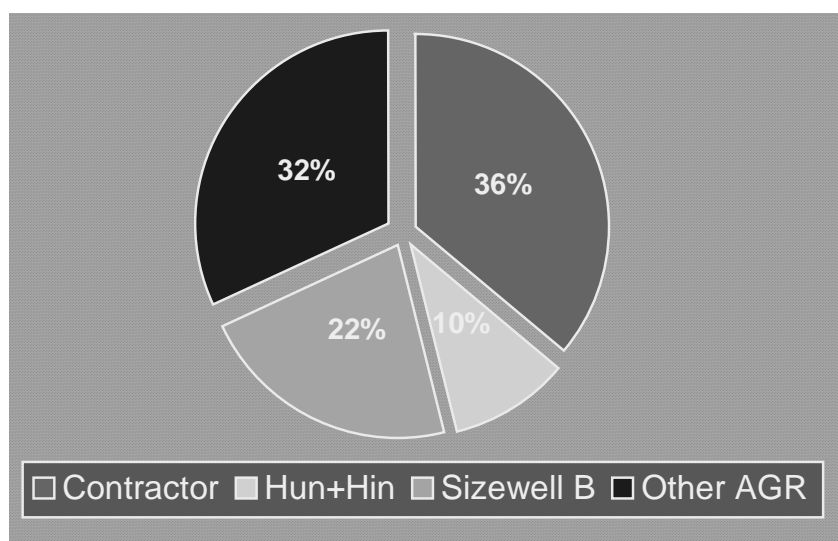
In 1999 the BEG outage team focused their thinking on the planned work. However, there was a need to react to emergent work discovered by the in-vessel inspections. This resulted in a combined in-vessel collective dose of 880 man·mSv, where 18 vessel entrants exceeded the project target individual dose of 10 mSv but not the CDRL at that time being 15 mSv. As a consequence, for the year 2000 outages the team decided to plan for both planned and possible emergent work resulting from the inspection campaign. Due to the nature of the work, and the environmental conditions under which it is conducted, the available resource is limited. It was considered that the scale of both the planned and emergent work would make it extremely difficult to remain compliant with the BEG CDRL.

As a contingency, the contractor in conjunction with the outage management team provided a dose management document for the work. This predicted a collective dose of 1053 man·mSv and a number of staff could be expected to exceed the CDRL. Additionally a mechanism was needed to address the impact of the Hunterston outage on the subsequent Hinkley outage, to be conducted by the same team and the possible issues arising from restrictions to employing contractor staff at BEG sites during the remainder of the year.

On the basis of the predicted dose from the above-mentioned document, HSED made a forecast, that the BEG Company collective dose would be 2.9 man·Sv (see figure 3). The BEG Company dose target for the year being 0.2 man·Sv per reactor (aggregate of 3 man·Sv).

HSED then performed a review comparing and scrutinising both the dose predictions and the strategy for various options of defect repair techniques. This review found the predictions to be pessimistic when compared to realistic and historic information. HSED suggested a total collective dose target of 650 man·mSv (cf. 1053) and indicated that the maximum dose to an individual should be imposed well under the CDRL. The foremost reason being a realistic assumption of average dose per entry to the vessel. HSED considered, based upon past outages and history of previously found defects on these reactors, that it was unlikely that there would be a need to exceed the CDRL for planned and emergent work in both the intended outages for year 2000.

Figure 3. Dose forecast 2000



As required under BEG Safety Rules, both Hunterston and Hinkley prepared pre-work ALARP assessments reports that underwent the required review by HSED prior to work commencing. Both sites were required to prepare review reports following outage completion. Following the Hunterston outage, Hinkley confirmed that, provided there was not a large degree of emergent work, they should not need to make a request to exceed the CDRL. Table 3 compares planned versus actual dose performance.

Table 3. Comparison of outage dose targets against actual for year 2000

Station	Planned collective dose constraint man·mSv*	Actual collective dose man·mSv	Planned individual dose constraint mSv	Actual maximum individual dose mSv
Hunterston	400	286	8	5.1
Hinkley Point	270	375	8	8.5
Sizewell	550	419	8	3.7

* Note that Hunterston and Hinkley only provided information for **planned** work, as they could not predict the extent of any emergent work, which is included within the **actual** totals.

Additionally as part of BEG self-regulation of practices, independent evaluations of the outage were performed by HSED at both sites and Sizewell B in year 2000. These evaluations identified good practices and possible areas for improvement; Together with local site reviews many of these were accepted as actions to improve for subsequent in-vessel and dose management campaigns.

The outcome (661 man·mSv) in Table 3 closely agrees with that suggested earlier by HSED (650 man·mSv) and was a significant improvement on 1999 collective (880 man·mSv) and individual

doses. Credit for this result is due both to the BEG sites and the contractor for learning from previous experience and for applying diligent dose management programmes. At the end of the programmed outage work in year 2000 doses were adequately planned and maintained below the CDRL.

Plant failures

In May 2000 the non-outage reactor at Hunterston B experienced a failure of a boiler tube and was returned to service at reduced output as a consequence. In November, after the BEG statutory outages were complete, the reactor was taken out of service to repair the tube and any other tubes exhibiting indications of wear. Contingency plans were made for limited inspection and to repair up to 10 tubes, dose estimates were made beforehand that predicted this work should be completed within the CDRL.

Before the return to service in May, BEG had assured the regulator that a programme of inspections to assess the full extent of the damage and effect repair to the tubes would be undertaken at the earliest practicable opportunity. The principle justification for the then planned vessel entry was therefore a requirement to return the reactor to full compliance with the safety case. Secondary justifications for vessel entry was the requirement to demonstrate compliance with the gaseous discharge authorisation, requiring “best practicable means” to be used to minimise discharges. A significant proportion of a discharge limit (S^{35}) was released in May as a result to protect reactor internal components and another similar discharge would be undesirable. Also it would be an opportunity to recover boiler surface area by repairs to economiser swage pieces and superheater bifurcation for continuous operation.

Inspections revealed that the number of tubes requiring repair was greater than anticipated, in excess of 80 tubes required replacement and early dose estimates of 2.5 man·Sv were estimated to complete the task. The available specialist manpower for in-vessel work was finite and could not be increased as it was not possible to train the necessary additional resource, especially in the time scale required to complete repairs without BEG forfeiting a significant commercial loss. It was decided to use the same contractor to perform the work. This contractor offered the benefit of having relevant recent experience and proven good improvement in dose management. The disadvantage was clearly that many of them, due to this special circumstance, would evidently exceed the CDRL for year 2000. The HSED Director was subsequently requested to authorise individual exposure above the CDRL with the agreement of the contractor. This was provided following consultation with the Company RPA and by issuing specific conditions.

The work programme would extend well into year 2001 covering four identified phases of work, supported by ALARP assessment documents all furnished to the HSED Director for review. The normal (vessel) top entry for all the work would increase the dose to nearly 3 man·Sv and a decision was made to perform bottom vessel entry by removal of a gas circulator to reduce the overall dose further. Benefits include improved safety to staff and their psychological well being, greater flexibility in staff selection and team composition, better quality control and reduced training requirements. These arrangements conserve individual dose by improved flexibility in dose sharing. In this case the risks to personnel from the work were considered to be acceptable when compared with the safety and financial benefits to society that were expected. The precautions taken and benefits expected more than comply with guidance [3] on the monetary value of dose.

Radiation protection specialists from the regulator and HSED undertook inspections to confirm that the work was indeed justified and ALARP. Specific meetings were held with the

regulator, safety committees and contractor representatives to explain the work programme and assure staff that safety was fundamental.

Hunterston B Economiser outage

This work was divided into four phases; phases 1-3 were undertaken in 2000/1 with a total dose of 474 man-mSv. Phase 4, undertaken in 2001 resulted in 833 man-mSv additional dose; the highest individual dose was 8.6 mSv where only 3 exceeded the very challenging individual dose constraint of 7.5 mSv, all notably below the CDRL. The alternative access route to the vessel through a gas circulator casing resulted in a saving exceeding 500 man-mSv than by top entries in “hot suits”. It provided additional improvements in productivity and reduced rework and for responding to emergent work. Alternative work management methodologies operated by the contractor also resulted in substantially reduced time and dose. The substantial difference between predicted and actual dose (2500 cf. 1307 man-mSv) was due to the work being performed noticeably quicker in much lower than expected dose rates at the sub-annulus.

Hinkley mini outage

The Hinkley non-outage reactor underwent a mini-outage vessel entry programme in December 2000 to allow inspection of its economiser boiler tubes. Also, additional work included the repair of a tube leak that occurred during the shutdown, plus repairs to recover defective boiler tubes and subheaders. In order to support the design safety case limits and return the station to nominal design output again, it was necessary to utilise individuals that had already exceeded and others who were near to exceeding the CDRL. The requirement to exceed the CDRL was subject to prior sanction from the Director, HSED. Although the predicted collective dose was 93 man-mSv the actual collective dose was 86 man-mSv.

BEG dose performance in years	Collective dose man-mSv		
	1999	2000	2001
Hunterston B collective dose man-mSv	446	788	961
Hinkley Point B collective dose man-mSv	675	638	103
Sizewell B collective dose man-mSv	786	499	271
BEG Company collective dose man-mSv	2876	2659	1781

It transpired that in the year 2000 a **total of 39** contract staff (from an overall 130 who participated in vessel entries) **exceeded the BEG CDRL**, 33 of these exceeded the level whilst working at Hunterston and the remaining 6 whilst at Hinkley. The highest individual (site) dose at Hunterston was 11.9 mSv and 13.8 mSv at Hinkley for the year. The highest individual dose at Hunterston in 2001 was 8.6 mSv.

As required in the BEG safety rules an investigation report has been provided for all contractor individuals who exceeded the CDRL (even though their employer’s investigation level is 15 mSv) and this report drew upon the pre-work ALARP assessment reports and post outage reviews.

Lessons learned from the CDRL sanctioning process

The introduction of a CDRL in a self-regulated organisation has been effective in reducing both collective and individual dose, coupled with sanction by the Organisations' Executive Director of Health & Safety, who is supported by a unit that undertakes independent review and evaluation.

The benefit of having a CDRL of 10 mSv provided motivation for enhanced dose reduction practices that resulted in minimising both individual and collective dose. The prominence of the CDRL produced a greater awareness for effectual dose management across both BEG and contractor staff alike. Post task and outage ALARP reviews have been invaluable in identifying areas and actions necessary for the improvement of dose reduction practices.

Preceding systems for the monitoring of radiation dose records for CDRL performance across BEG placed a sizeable burden on site Health Physics services staff. Recent changes to adopting the Electronic Personal Dosimeter as the legal dosimeter and consolidating approved dosimetry services has now improved matters and is essential to the process.

It has been a common practice to plan the engineering programme and subsequently perform the ALARP assessment. At this point the work programme is often fixed and difficult to change. Some important ALARP considerations are required for management decision before possible revisions to the work programme. Hence, ALARP assessments are now integrated into the overall project plan and highlighted early on whilst tendering and co-operating with our contractors in outage projects.

In view of the plant problems experienced and the anticipated increase in dose rates with continued operation of the reactors, the strategy for future in-vessel inspection and repairs and the impact of the Company Dose Restriction Level needs has been reviewed. One important initiative at Hinkley Point is for enhanced vessel inspection in outage year's 2002/3 to enable a case to be made to the Inspector for remote in vessel inspection in subsequent years.

References

- [1] Statutory Instruments 1999, No.3232, The Ionising Radiations Regulations 1999, The Stationery office. ISBN 0-11-085614-7.
- [2] British Energy web site: www.british-energy.com
- [3] Documents of the NRPB Occupational, Public and Medical Exposure, volume 4, No.2 1993.

COMMONALITY INITIATIVES IN US NUCLEAR POWER PLANTS TO IMPROVE RADIATION PROTECTION CULTURE AND WORKER EFFICIENCY

W. Wood

American Electric Power, USA

D. Miller

Donald C. Cook Nuclear Power Station, USA

Abstract

Many US nuclear power plants have learned that common procedures, policies, instrumentation, tools and work practices achieve improvements to the radiation protection culture. Significant worker efficiency achievements are accomplished especially during refueling outages. This paper discusses commonality initiatives currently being implemented at many US plants to address management challenges presented by deregulation of the US electric industry, reduction in the pool of outage contractors and aging of the experienced radiation worker population. The new INPO 2005 dose goals of 650 person-mSv/year for PWRs & 1 200 person-mSv/yr for BWRs will require new approaches to radiation protection management to achieve these challenging goals by 2005.

Introduction

This paper describes the 5-year plan developed to achieve the 2005 INPO dose goals for the D.C. Cook Nuclear Power Station. Due to page limits on the workshop papers, please see the handout slides for discussion of the commonality initiatives at US nuclear power plants. The D.C. Cook 5 Year Plan is provided below:

Executive summary

This 5-year plan summarises the existing programs for dose reduction. It also outlines exposure reduction activities to be implemented. Significant components among these activities are: installation of permanent shielding on the NRV/QRV walls, the removal of RTD lines, and the incorporation of PEX Resin into the CVCS beds. This document provides a vehicle to prioritise and direct these initiatives.

Introduction

DC Cook Nuclear Power Plant is committed to minimising radiation exposure to meet PWR INPO first quartile performance and to be recognised as one of the superior radiation protection programs in the nuclear power industry.

The vision: DC Cook becomes a world class performer in nuclear power generation.

This vision will be realised through low collective dose to the workforce, low dose to the general public due to plant operations, low generation of radioactive waste, and a high degree of regulatory performance.

The 5-year dose reduction plan is the road map to superior ALARA performance. It will be periodically reviewed and revised as the station and industry demands change. On an annual basis, the ALARA Committee will review the results of the plan and recommend changes as warranted based on conditions and requirements present during that time period. This should occur prior to the start of the next calendar year.

Statement of purpose

One of the fundamental principles of radiation protection is that radiation exposures be kept as low as reasonable achievable (ALARA). Furthermore, the Nuclear Regulatory Commission, the Institute of Nuclear Power Operations, American Nuclear Insurers, and others use cumulative radiation exposure as an indicator of nuclear plant performance. DC Cook is committed to maintaining collective radiation exposure among the lowest in the country.

The primary purpose of the DC Cook 5-year Dose Reduction Plan is to establish exposure reduction activities to be implemented which will help achieve dose reductions; projecting us into the first quartile of dose performers.

This will be accomplished through an aggressive approach to ALARA. Radiation sources will be reduced through filtration, flushing, system chemistry controls, decontamination, cobalt reduction, and eliminating certain high dose components. Where the source cannot be removed, temporary and permanent shielding will be used when dose effective. Reductions in collective outage dose will be realised through shorter and more efficient outages. A highly trained and motivated work force, utilising the latest technology, will also make a significant contribution to lowering personnel radiation exposure during both outages and non-outage times.

This report also serves as a status report for on-going exposure reduction actions and as a summary of exposure reduction activities.

Plant dose history and projection (in person-rem)

The table below shows DC Cook's dose history and projections, and shows the gap we must bridge to reach median and upper quartile collective radiation dose values.

Year	Cook Annual	Cook 3 year average	Cook delta to median	Industry Median	Cook delta To 1 st Quartile	1 st Quartile 3 year average
1999	171	141/unit	+43	98	+63	78
2000	338	105/unit	+23	82	+26	79
2001	27	89/unit	+2	87	+13	76
2002	250	102/unit	+25	77	+32	70
2003	240	86/unit	+13	73	+24	64
2004	6	82/unit	+13	69	+24	58
2005	114	60/unit	-5	65	-8	52
2006	108	38/unit	-22	60	-7	45

Shaded areas are forecasted numbers. See below for breakdown of Cook Annual dose projections.

2002 Dose (250 person-rem)

- U2C13 – 110 person-rem;
- U1C18 – 90 person-rem;
- CRDM inspection – 15 person-rem each unit (30 total);
- Non-outage dose – 20 person-rem (routine and DCP work).

2003 Dose (240 person-rem)

- U2C14 – 90 person-rem;
- U1C19 – 88 person-rem;
- Removal of RTD lines – 20 person-rem each unit (40 total);
- Installation of S/G platforms – 6 person-rem each unit (12 total);
- DCP work – 2 person-rem;
- Non-outage dose – 8 person-rem.

Outage dose reductions will be achieved through the use of PRC-01 resin, zinc injection, and outage efficiencies. Non-outage dose reductions will be achieved through optimization of preventive maintenance schedules (including radiological survey schedules), use of remote monitoring equipment, and continued improvements in worker efficiencies.

2004 Dose (6 person-rem)

- Non-outage dose – 6 person-rem
- No outages
- No design change dose

A reduction in the annual non-outage dose will be attained by means of a continued optimisation of preventive maintenance schedules and remote monitoring.

2005 Dose (114 person-rem)

- U2C15 – 54 person-rem;
- U1C20 – 54 person-rem;
- Non-outage dose – 6 person-rem.

A reduction in outage dose will be realised by the removal of RTD lines and continued benefit from the use of zinc injection and PRC-01 resin (reducing source term).

2006 Dose (108 person-rem)

- U2C16 – 51 person-rem;
- U1C21 – 51 person-rem;
- Non-outage dose – 6 person-rem.

Estimating approximately a 5% reduction attributed to the continued use of zinc injection and PRC-01 resin along with continued efficiencies in remote monitoring, preventive maintenance scheduling and radiation worker practices.

Administrative ALARA program

ALARA procedures

The two governing ALARA procedures, PMP 6010.ALA.001, ALARA Program – Review of Plant Work Activities, and PMP 6010.ALA.002, ALARA Committees were recently revised to increase their effectiveness. Enhancements included clarification of responsibilities, lowering the dose thresholds for reviews and the creation of three data sheets that strengthen our review of Design Change Packages, ALARA In-Progress reviews, and ALARA Suggestions. It also clarified management's ownership of dose budgets and dose reduction, and included a section for source term reduction.

Exposure limits

Minimising exposure at the individual level is an essential part of reducing the collective site radiation exposure. It is our policy to not just maintain personnel exposures below regulatory and administrative limits, but to keep individual exposure ALARA. In an effort to maintain individual occupational exposures as low as reasonably achievable, a site administrative limit has been established to maintain personnel radiation exposure to <2 000 mrem during any year. This limit also helps to encourage departments to equalise dose among their work force.

Training

Initial Radworker Training, and the annual Radworker requalification, stress ALARA awareness and the potential for changing or abnormal radiological conditions; including the actions required when these conditions occur. This training not only increases worker awareness of radiological conditions; it also reduces their dependence upon the radiation protection technicians in the field. Radiation Protection Department staff attend on and offsite training, seminars, and perform industry benchmarking to keep abreast of new developments in the industry and their impact on our site.

ALARA committees

DC Cook has 2 ALARA Committees, the ALARA Committee and the ALARA Sub-Committee. The ALARA Committee is chaired by the Plant Manager and is composed of

departmental Directors, Managers, and Superintendents. The ALARA Sub-Committee is chaired by the RP Supervisor – ALARA and has representatives from the different departments.

Cook Plant procedure PMP-6010.ALA.001, ALARA Program – Review of Plant Work Activities, requires jobs estimated at ≥ 1 person-rem TEDE to receive an ALARA Review by the ALARA Sub-Committee for establishment of techniques and requirements to aid in maintaining exposure ALARA. The ALARA Committee reviews all jobs estimated to be ≥ 5 person-rem TEDE.

Departmental dose reduction plans

Departments that expect to receive greater than 100 mrem in any year are required to prepare a Dose Reduction Plan outlining the actions they intend to implement during that year to reduce the dose received by personnel in their department. The requirements for these plans are in the ALARA Procedure PMP-6010.ALA.001.

Shielding

To reduce personnel exposure, we install approximately 49 000 pounds of shielding (~25 shielding packages) during a typical refueling outage. The ALARA review/job planning process assesses the use of shielding for jobs during outage and nonoutage. Shielding materials include lead blankets, lead bricks, and water shields.

Hot spot tracking

Hot Spots are defined in the Hot Spot procedure, 12-THP-6010.RPP.013, as an accessible component having a contact dose rate ≥ 100 mr/hr and five times the general area dose rate at 30 cm.

Each identified Hot Spot is entered and tracked in the Hot Spot Tracking Log. The Hot Spot is assigned a survey frequency to monitor for changes in location or activity. A Hot Spot Evaluation Worksheet is also completed for every Hot Spot. If the work sheet shows that removal or flushing is cost beneficial, then a Work Order request is generated for its removal.

ALARA cost benefit analysis

The current value associated to 1 person-rem of exposure is \$18 154. This value is used to aid in justifying design changes, modifications, and other major expenditures. This figure was established in 1989 and was reevaluated again in 1995.

Benchmarking

Radiation Protection personnel maintain contact with their counterparts at other nuclear utilities to exchange information and ideas. Efforts are made to attend regular meetings such as the Region III Radiation Protection Managers, PWR/RP ALARA Committee, Health Physics Society, EPRI, INPO, NEI, and others. Information exchanged is used to continually improve and maintain knowledge of industry changes and innovations related to exposure reduction. Evaluation of new

products such as robotic technology, computer surveys and remote monitoring are essential for continuing improvement in radiation protection.

The corrective action program is used to track good practices from benchmarking which merit evaluation for implementation. Some of the items implemented from past benchmarking include:

- The new ALARA Suggestion cards.
- Formalised guide for ALARA reviews of procedures and design changes.
- Radiological Risk Significance categories (A, B, or C) for RWPs.
- Changes to the ALARA procedure to perform “in-process” ALARA reviews for high dose/high risk significance jobs or projects.
- RP Turnover Sheet to use for days to gives to nights (front side) then nights gives back to days (back side of paper). Sections include: Posting Changes, Condition Reports Written, work summary, look ahead, relief required, and expectations for nightshift/dayshift.
- Radiological Pre-job plans (i.e. cavity decon).
- Re-established area supervisors and leads with adequate staffing.
- DOP testing of HEPA ventilation units and HEPA vacuums.
- Changes to High Radiation and Locked High Radiation area signs (new signs have a visual and dynamic difference).
- Reinstatement of outage handbooks (U2C13).

Source term reduction (STR)

Source Term refers to activity in piping or other plant components which, through various processes, can be removed. The removal of this source term results in lower exposure rates to radiation workers and to reduced “Dose to the Public.”

STR to be implemented prior to U2C13:

- Flushing of the RHR System – the RHR water will be flushed to the CVCS HUT using RCS pressure, prior to placing the RHR into service. This will reduce the increase in RCS activity normally seen during shutdown, because the RHR system will be at the same temperature and oxygen content as the RCS prior to being placed into service.
- Reduced activity in the reactor cavity – Filling the reactor cavity using the RHR’s Hot Leg injection point instead of the Cold Leg injection point will reduce the activity in the reactor cavity because the fill will not flow through the reactor core.
- RCS Vacuum Fill and Vent – Reduces the RCS fill time by 6 hours and eliminates RCP sweeps which will reduce RCP seal wear and associated maintenance resulting in reduced dose.
- Fuel assembly modifications – The unit will be refueled using 80 fuel assemblies containing ZIRLO™ cladding, guide thimbles, instrument tubes, and mid-grids. ZIRLO™ is a zirconium-based alloy that enhances fuel reliability and achieves extended fuel burn-up. The thimble plugs will also be removed from all the fuel assemblies allowing for simpler refueling processes; reducing radiation exposure.

Source term reduction team

We have assembled a Source Term Reduction Team with members from RP, ALARA, Operations, Maintenance, Environmental, Chemistry, and Engineering to evaluate and prioritize source term reduction processes. Processes include hot spot removal, flushing, cobalt reduction and shutdown chemistry.

Cobalt reduction

Particulate cobalt entering the RCS is activated as it passes through the reactor. Cobalt isotopes are the major contributors of exposure at the Cook Plant. Reduction of radioactive cobalt precursors, such as ⁶⁰Co, in the primary system will result in decreased dose. As a matter of perspective, one gram of ⁶⁰Co has a specific activity of >1 000 curies.

The ALARA Group using the Data Sheet 3, ALARA Review form found in the ALARA procedure (PMP-6010.ALA.001), reviews plant modifications, including all Design Change Packages. Data Sheet 3 guides the reviewer to look for and eliminate ⁶⁰Co. Plant Engineering procedures and training also discuss reduction of cobalt bearing materials.

Some important contributions to cobalt reduction have already been accomplished. The steam generators in both units have been replaced with generators having lower cobalt content than the original design. Inconel fuel assembly grid spacers were replaced with zircaloy spacers in both units during refueling outages in 1990. A cobalt reduction program database that identifies cobalt-bearing valves in communication with the primary system was developed in 1992. Since the development of this database, valve replacements and valve replacement parts have been ordered cobalt free by the warehouse.

Chemistry program enhancements

Power Operations – Moving from a “modified” RCS lithium program to a “coordinated” program. The Chemistry Department is funding a system materials evaluation by Westinghouse that will allow higher coolant lithium concentrations at the beginning of each cycle. This change will allow lithium concentrations to be controlled at levels high enough to produce a constant pH throughout the cycle, minimising early cycle corrosion product deposition on fuel surfaces and subsequent transport out of the reactor core.

Startup Chemistry – Continuing to follow a strategy to minimise time spent with acid reducing condition during RCS heatup. The aim of this strategy is to avoid creating a chemistry regimen that would de-stabilise core deposits and promote transport around the reactor coolant system.

Shutdown Chemistry – Revising reactor coolant de-lithiation practices to achieve acid conditions early in cooldown process. The coolant will be de-lithiated during the downpower at a rate that will result in an at-temperature pH of 6.5 at entry into Mode 3. Also, revising the RCS degas process to ensure that sufficient dissolved hydrogen is retained in the coolant to provide an adequate margin for maintaining corrosion products in a soluble form.

Foreign material exclusion program

The Plant Manager Procedure for FME contains guidance on cleanliness when working on valve internals with emphasis on thorough cleaning to ensure no loose cobalt/stellite is left inside the valve that could later get into the reactor. The FME program also controls debris/foreign materials from entering plant system, which prevents that material from damaging the fuel, or other components.

Worker productivity enhancements

ALARA work planning

ALARA Job packages are used in pre-job planning and include lessons learned from past work. By reviewing past experiences, we can usually find ways to do the work more efficiently, reducing the overall personnel dose. The ALARA group breaks jobs into smaller activities to see where additional dose can be saved, if a mock-up would increase efficiency, or if other facilities have something helpful to share. ALARA dose goals and incentives are also used to motivate workers to work more efficiently.

ALARA in-progress and post job reviews

The ALARA group performs in-progress reviews, at 50% and 80% of the estimated dose, of all jobs with an estimated dose budget greater than 1 Person-rem. Post-work reviews are also performed after these jobs are complete. These reviews are placed into the ALARA Job package for review and incorporation of lessons learned prior to performing the work again.

ALARA suggestion program

An individual or work group may submit suggestions to improve how a job is performed.

Since the rebirth of the Suggestion/Incentive program, in early 2000, an average of 10 suggestions have been received per month. The estimated dose savings from these suggestions is 5 person-Rem, or ~\$90 000 (\$18k/person-Rem).

Initiation of an ALARA parking spot has helped to increase interest in the ALARA Suggestion Program. The parking spot is used to reward personnel who submit suggestions resulting in large dose savings.

Mock-ups

Mock-up activities are performed to help personnel become more efficient in their task and to identify areas needing improvement, especially new technologies or first time evolutions. By performing the task in a non-radiological environment, we can experiment with new ideas while becoming proficient with the job at hand. Examples of mock-ups which have aided in dose reduction include: RCP work, S/G activities, Reactor Vessel Support work/cleaning, filter change-outs, non-routine valve work, cutting and machining of RCS piping (SGRP), RCS pipe end decon (SGRP),

Steam Generator girth cut (SGRP), moisture carry over testing, and many other new or unfamiliar tools, processes, and technologies.

Exposure trending

Comparison of dose received for prior evolutions allows for challenging dose goals to be established. When the ALARA dose goal are met, ALARA incentives awards are often given in recognition of good ALARA work practices.

Contaminated square footage

Many areas of the plant that were once posted as contaminated areas have been decontaminated and are maintained as clean areas. This enhances access while reducing contaminated clothing and DAW generation. The plant enforces a clean as you go policy for all workers. Contaminated area square footage is tracked by the RP Department and is an indicator of radiological performance.

Effective exposure reduction practices

Design changes/system enhancements

Design Change Packages are reviewed by the ALARA group, using the Data Sheet 3, DCP ALARA Review form in the ALARA Program procedure (PMP-6010.ALA.001). The Data Sheet 3 is a check-off list with 117 questions prompting the reviewer to look for exposure savings design criteria. The questions are divided into 12 sections, including shielding, source term reduction, contamination control, and system layout.

Remote monitoring

Remote monitoring is used to reduce the exposure of personnel performing inspections or walk-downs, and for RP technicians providing job coverage. The individual is able to remain in a low dose area while work is being performed in an area of greater radiological significance. Remote monitoring is used for various refuelling, RCP, and Steam Generator activities. Radiation Protection also uses robotics (e.g. under water subs, and the scavenger) to save personnel dose.

Remote technology is an important tool for reducing personnel dose. Use of remote monitoring and robotics has greatly increased throughout the nuclear industry. Several actions to evaluate and implement additional remote technologies are listed in this 5 year plan.

Exposure reduction techniques being investigated for implementation 2002-2006 (within the next five years)

Project	Benefit	Owner	Due	Priority
Removal of U2 RTD lines RPA 5193	Reduce exposure for future outages	B. Story	2003	1
Removal of U1 RTD lines RPA 5006	Reduce exposure for future outages	B. Story	2003	1a
Zinc Injection	Reduces radioactive material in the RCS	S. Griffin / D. Kozin	2003	2
PRC-01 Resin use in U-1	Reduces RCS source term and contamination levels	S. Griffin / D. Kozin	2001	3
Installation of Jib cranes in U2 CTMT @ 22&23	Reduces outage dose by decreasing time and exposure during mobilisation and demob.	B. Story	2003	4
Installation of Jib cranes in U1 CTMT @ 12&13	Reduces outage dose by decreasing time and exposure during mobilisation and demob.	B. Story	2003	4a
Improve turbine side access control	Increase RP interaction with rad workers	R. LaBurn	2002	5
Improvements to LHRA barrier @ Accumulators	Reduce exposure by eliminating RP constant coverage	R. LaBurn	2002	6
Install permanent S/G Platforms in U-2 CTMT RPA 5195	Reduced exposure future outages	B. Story	2005	7
Install permanent S/G Platforms in U-1 CTMT RPA 5194	Reduced exposure future outages	B. Story	2005	7a
Install permanent shielding @ NRV/QRVs	Reduce exposure future outages	L. Green	2003	8
Storage of lead in CTMT and Auxiliary buildings	Reduce exposure & time associated with transportation of shielding	L. Green	2003	9
CTMT penetrations for RP communications	Increase use of remote monitoring	R. LaBurn	2005	10
CPN-1 quick removal 4 bolt modification to Blind Flange	Save time and dose in lower CTMT	K. Rolins	2003	11
Improvements to 587' Drumming Room ventilation	Increase ventilation allowing processing of highly contaminated waste	R. Hedgepeth	2004	12

Project	Benefit	Owner	Due	Priority
Permanent shielding in the Regen and Excess Letdown Hx. Rooms	Saves exposure from installation and removal in future outages	L. Green	2004	13
Flushing of Hot Spots during U2C13	Reduce exposure in plant areas	S. Griffin	2003	14
Flushing of Hot Spots during U1C18	Reduce exposure in plant areas	S. Griffin	2003	15
Pre-approved installation specifications for temporary shielding	Save engineering time and decrease time for installation – simplify the use of temporary shielding	L. Green	2003	16
Storage of scaffolding in containment during power operations	Reduced time and exposure for scaffold installation and removal	T.Tillstrom	2003	17
Cobalt Reduction Program for replacement of valves with high cobalt content (Evaluate the 1992 document)	Reduce the cobalt in the RCS	S. Griffin / Design Engineering	2002	18
Mock-up facility in the Training building	Increase worker efficiency, reducing dose	TBD	TBD	19

**INDUSTRIAL ACCIDENTS IN RADIOLOGICAL CONTROLLED AREAS:
THE IMPORTANCE OF RADIATION PROTECTION
IN THE ORGANISATION OF THE EMERGENCY AID**

J-L. Fenolland

EDF Golfech NPP, France

E. Laporte

EDF Saint-Alban – Saint Maurice NPP, France

Introduction

After some disappointments when the first French nuclear units were started, it became clear that all the aspects linked to radio-protection needed to be taken into account in the context of emergency aid in the case of an industrial accident in radiological controlled area.

Context

In the case of an accident involving people, on-site first aid is provided by permanent services of the power plant. These teams are trained in first aid and fire-fighting. They are well trained in radioprotection.

The specificity of an industrial event in controlled zone is that the victims' conventional injuries, whether it be a wound, a burn or a fracture, can be complicated by radioactive contamination.

If it is justified, the exterior emergency services (firemen and medical teams) are immediately called in. These teams are not necessarily trained in radioprotection.

Stakes

The main aim is to have no delay in the medical care of the victims. It is therefore very important that the exterior emergency services are not perturbed by the radiological conditions of the operation. Anxious state of mind in an unknown situation can generate unsuitable behaviour.

In parallel, radioprotection must be provided for the victims and the teams who are assisting them.

Organisation

In case of accident in radiological controlled area, a specific kind of organisation needs to be set up which depends on the number of casualties and the seriousness of their injuries.

When the usual means of first-aid are insufficient and there is a lack of balance between the available resources and the needs, an Emergency Plan with a Sanitary approach needs to be implemented, which places the same means at our disposal, by adapting them to the situation, as those which are provided in the case of events which affect the safety of installations.

The events successively take place on the scene of the accident, then in an intermediate structure which is for grading the casualties and for serious medical intervention, called Medical Outpost Station and finally in the services of the different hospitals.

The aim of this organisation is to make a list of all the casualties, to take care of the most seriously injured as soon as possible and to optimise the evacuation of the victims.

The role of the medical teams and the radio-protection team on site is to ensure and to deal with the radio-protection of the victims and the first-aid workers throughout their intervention, from the NPP to the evacuation hospital.

Concerning the radio-protection and whatever the organisation used, the main principles to set up are always the same:

- the evaluation of the radiological risk;
- the marking of the sites which have a high radiation area, and the contaminated areas;
- the optimisation of the radiological exposure suffered by the casualties and the first-aid workers;
- the availability of individual protections (overalls, breathing protection, etc...) and collective protections;
- the control of the measures taken and their efficiency;
- the recording of the radiological conditions that have been sustained, in terms of both external and internal exposure, whether by inhalation, ingestion or contaminated wound.

The casualties are transported from the site of the accident to the Medical Outpost Station, which is put into place if the situation justifies it. It is ideally a large, open and sheltered area, situated away from the contaminated areas. The medical and radiological grading is carried out here. It is essential to remember that, in every case, standard medical emergency takes precedence over the radiological risks.

In the Medical Outpost Station, two distinct zones are defined: the conventional zone and the contaminated zone.

When the condition of the victim permits, the following gestures are carried out concerning radiological risks:

- undressing, which contributes decreasing the contamination of a factor 10;
- external decontamination by washing;
- total body counter measuring the internal contamination.

In the Medical Outpost Station, the radio-protection team's mission is to manage the contaminated zones, to inform the participants at the entrance of the zones with a contamination risk, to control everybody and everything at the exit, and finally to deal with the waste.

The casualties are then sent to hospital. The radioprotection must be ensured during the transport, and the personnel and equipment must be controlled at the end of the mission.

From the point of view of radioprotection, the reception of a contaminated casualty must respect the same principles as classical French theatre:

- the principle of unity of place; the concern of reducing the spreading of the contamination inside the hospital;
- the principle of unity of time, which means decontaminating the casualties as soon as possible;
- the principle of unity of action, which consists to work in close collaboration both the hospital staff, who are dealing with the casualties, and the radiological teams, who are dealing with the radiological risks.

If these few rules are followed, they ensure the same quality of treatment for the casualties who are contaminated as for the conventional casualties.

Conclusion

Even if a serious accident with a large number of casualties is highly unlikely, an industrial event with one contaminated victim can happen at any moment and these situations need to be well prepared to be dealt efficiently.

The keys to success are:

- the training of the fire-fighting and medical staff who intervene on the site;
- radio-protection support from the power plant teams throughout the rescue operation;
- regular drills with role-plays which will test the different interfaces, including the reception in the hospital services.

Finally, a relationship of trust absolutely needs to be worked on and maintained with the different partners who take part in those kind of accident situations.

HEALTH PHYSICS SELF-ASSESSMENT AND THE NUCLEAR REGULATORY OVERSIGHT PROCESS AT A NUCLEAR POWER PLANT

R.S. Schofield

Health Physics Division, Southern California Edison Company, USA

Abstract

The U.S. Nuclear Regulatory Commission has developed improvements in their Nuclear Power Plant inspection, assessment and enforcement practices. The objective of these changes was to link regulatory action with power plant performance through a risk-informed process which is intended to enhance objectivity. One of the Strategic Performance Areas of focus by the U.S. NRC is radiation safety. Two cornerstones, Occupational Radiation Safety and Public Radiation Safety, make up this area. These cornerstones are being evaluated through U.S. NRC Performance Indicators (PI) and baseline site inspections.

Key to the U.S. NRC's oversight program is the ability of the licensee to implement a self-assessment program which proactively identifies potential problems and develops improvements to enhance management's effectiveness. The Health Physics Self-Assessment Program at San Onofre Nuclear Generating Station (SONGS) identifies radiation protection-related weakness or negative trends. The intended end result is improved performance through rapid problem identification, timely evaluation, corrective action and follow-up effectiveness reviews.

A review of the radiation protection oversight process and the SONGS Health Physics Self-Assessment Program will be presented. Lessons learned and management tools, which evaluate workforce and Health Physics (HP) staff performance to improve radiological practices, are discussed.

U.S. nuclear regulatory oversight program background

The U.S. NRC has developed a more objective process for assessing a Nuclear Power Plant licensee's regulatory and safety performance (NEI, March 2000). This new process uses a risk-informed work process and focuses on three Strategic Performance Areas:

- reactor safety;
- radiation safety;
- safeguards.

Seven corner stones were identified by the NRC to capture these three strategic areas for which Performance Indicators (PIs) were established to monitor and assess a licensee's performance, Figure 1. U.S. NRC inspection modules are used to supplement the performance indicators. The

overall mission of the U.S. NRC in the implementation of these PIs is based upon protecting the health and safety of the public and occupational radiation workers.

Radiation safety cornerstone and performance indicators

The Radiation Safety Cornerstones are broken down into two areas: Occupational Radiation Safety and Public Radiation Exposure. Performance Indicators were established for each of these areas. The Performance Indicators were selected based upon their ability to provide an objective measure and be readily identifiable based on industry experience. The dose criteria employed does not represent levels of dose that are risk significant and represent criteria generally at or below U.S. NRC regulations. These PIs seek to protect the health and safety of workers involved with exposure from licensed and unlicensed radioactive materials during routine operations at nuclear power plants, Figure 2.

The Occupational Radiation Safety performance indicators are:

- Technical specification high radiation area (>0.01 Sv/hr, >1 Rem/hr) occurrences.
- Very high radiation area occurrences.
- Unintended exposure occurrences.

The Performance Indicator is determined by summing the reported number of occurrences for each of the three above indicators over the previous four quarters for:

- a loss of radiological control over access or work activities within a high radiation area (>0.01 Sv per hour at 30 cm, >1 Rem/hr at 30 cm) or very high radiation area (>5 grays, in one hour at one meter, >500 Rads in one hour at one meter); or
- a degradation or failure of one or more safety barriers which results in an unintended occupational exposure equal to or exceeding;
- 0.001 Sv, (100 mrem) TEDE;
- 0.05 Sv, 5 rem CEDE;
- 0.015 Sv, (1.5 rem) LDE;
- 0.05 Sv, (5 rem) SDE;
- 0.001 Sv, (100 mrem) minors or declared pregnant worker;
- 0.50 Sv, (50 rem) SDE from a discrete radioactive particle.

The U.S. NRC's decision to move to a risk-informed regulatory program was in part driven by the U. S. nuclear power industry's maturation. The marked improvements in the nuclear industry's safety and reliability caused the U.S. NRC Commission to re-evaluate their inspection programs and the licensee's role to assess themselves.

SONGS self-assessment program

Self-Assessment has been a normal way of doing business at SONGS for a long time.

It was very centralized with the Nuclear Oversight Division (Quality Assurance, Internal Safety Evaluation Group) performing assessments and evaluations of other divisions.

Today, Self-Assessment at SONGS is driven by line management. The site has established Division Self-Assessment Leaders, a Self-Assessment Forum and a Self-Assessment Steering Committee, Figure 3 (SONGS, November 1998).

Divisional Self-Assessment leads focuses on internal performance and engage staff to participate in the program. The lead individuals apply common station self-assessment procedures and guidelines. They oversee the implementation and effectiveness of self-assessment activities within their division and participate in the station's Self-Assessment (SA) Forum.

The SONGS' SA forum mission is to lead, co-ordinate and promote a consistent station Self-Assessment approach. The forum consists of SA leads from each division. The Vice-President of Nuclear Generation is the Executive Sponsor. The forum meets quarterly and reviews, proposes or discusses Self-Assessment activities. The SONGS VP has challenged the Forum to engage employees, focus on results and become an industry leader. The Forum also takes direction from the station Self-Assessment Steering Committee.

The SA Steering Committee consists of senior managers who assess the overall effectiveness of the self-assessment activities and identify targets for improvement.

Health physics self-assessment program

The Health Physics Department Self-Assessment Program (HPD SA) complies with the Station SA elements outlined in Figure 4. It employs several unique work practices to assess Health Physics and radiation worker performance. As Figure 4 indicates, the station program consists of Planning, Implementation/Evaluation and Reporting.

HP self-assessment planning

The Health Physics Department Self-Assessment Planning (HPD SA) includes the scheduling of directed or focused assessments. Health Physics management reviews self-assessment identified problem areas. Work is scheduled collectively and HP management determines topics and team members to conduct self-assessments (audits) a year in advance. The HPD typically conducts four focused assessments annually.

Many of the HP employee training topics are selected based on SA identified performance issues. Each quarter, the HPD teams with the station Training Department and conducts operating or industry experience training where off-normal radiological events are discussed and examined for applicability at SONGS. These industry events are distributed by INPO via their INPO web page and the new ISOE Level 3 database.

Divisional annunciator panels with associated metrics are established to monitor fundamental work practices. These annunciators permit management to evaluate divisional performance against prescribed metrics for key Health Physics work activities.

Several approaches are employed to engage employees in SA activities. A Self-Assessment team composed of union and management employees meets approximately three times a quarter to

review performance results and recommend actions to improve performance. Team members rotate on an approximately 18-month basis. The team is responsible for communicating performance results and actions to HP staff as well.

These quarterly communications are conducted in a face-to-face manner during shift turnovers or management staff meetings by a union HP employee presents the SA report. An HPD web site is used to complement these HP SA presentations.

Station management encourages employees to participate in self-assessment by identifying and reporting problem areas or off-normal events. Employee's report such events in an electronic-based "action request" computer based system. The HPD initiates about 400 action requests per quarter.

HP management expectations are communicated in Divisional Performance Standards. Thirty-two HP standards have been prepared which capture divisional business practices. These standards are controlled and electronically available on the HP web site. These standards inform employees of their responsibilities beyond procedural or regulatory requirements. In this manner, management is positioned to hold employees accountable for their performance. Examples of performance standard subject matter includes: Health Physics posting, container labelling, reactor mode change radiological checklists, industrial safety and tailboard conduct. The HP SA program is responsible for the standard development and content.

HPD self-assessment implementation and evaluation

HP SA program implementation and evaluation consists of Employee Leadership Observations, Cause Evaluations, Precursor Trending, Focused Assessments, Industry Experience Reviews and Benchmarking, and External Assessment Reviews.

Employee leadership observations

HP management and voluntary union employees conduct monthly Leadership Observations (LOs). These observations call for employees to perform walkdowns in designated areas of the plant observing work and area conditions. Pre-established LO questions are answered by employees and submitted as an "action request". Leadership observations correct problems in the field, reinforce good worker practices, and serve as a trending mechanism to identify precursor problems. The Health Physics Department completes approximately 150 leadership observations a quarter.

Cause evaluations

Significant radiological events are investigated and evaluated for cause. Station procedures establish threshold criteria and requirements for the conduct of such an evaluation. Cause evaluations and corrective actions are captured in the "action request" system and discussed with Health Physics management leads. Significant results are captured as required reading on the HP web page. The Health Physics Department conducts approximately 5-10 cause evaluations per quarter.

Focused assessments

The Health Physics Department conducts approximately four radiological-focused assessments per year. These assessments are performed in accordance with station guidelines and documented in the “action request” system. Past assessment topics have included:

- radiation worker readiness for refuelling outages;
- health physics employee training effectiveness;
- health physics employee qualification manual work process;
- receipt of radioactive materials work process.

Additionally, the Health Physics Department participates in a U.S. NRC Region IV “round robin” industry peer radiation protection focused assessment of utilities. San Onofre Nuclear Generating Station’s (SONGS) Health Physics Department hosted an industry peer focused assessment in which an eight-member team (five industry peers) evaluated the station’s radiation protection work practices and performance. Results were informative and recommendations were presented to the SONGS Vice-President and site managers.

Industry experience

The HPD evaluates radiological industry events (Operating Experiences) routinely. Industry events are evaluated for applicability and corrective actions taken as necessary to minimize the likelihood of a similar occurrence at San Onofre Nuclear Generating Station. Significant events are communicated to employees as “required reading” on the HP web site. Some events are discussed in the quarterly Industry Experience training conducted by HP SA and the training division. Events are documented on the HP web site and used as lessons learned for radiological work planning and discussed in tailboards with site employees. The division evaluates approximately 10 industry events per quarter.

External assessment reviews

External radiation protection assessments such as audits or inspections from Nuclear Oversight (QA), the Nuclear Regulatory Commission, Institute of Nuclear Power (INPO), National Voluntary Laboratory Accreditation Program (NVLAP) or World Association of Nuclear Operations (WANO) are included in the Self Assessment program.

Trending precursor events

All of the above self-assessment activities are trended by the division using the “action request” (AR) system. Low level events which do not fall into the above categories are captured as simple trend “action requests” and collectively evaluated. Evaluations which identify commonalties or are viewed as continuing problem areas are investigated in further detail. The division initiates about 50 trend Action Requests per quarter.

Quarterly assessment and reporting

The HP SA team meets quarterly to review trends and assess divisional performance. Commonalties or significant precursor events are identified and recommendations made for further action. The radiation protection manager documents the HP SA team and recommendations and submits a quarterly report to the station Vice-President. As mentioned previously, the HP SA team communicates performance results and actions to staff members.

Health physics assessment of NRC performance indicators

An assessment of the Occupational Radiation Safety Cornerstone was completed by a Health Physics team. The assessment's objectives were to review programmatic controls and employee work practices/behaviors associated with the radiological PIs to minimize the likelihood of occurrence at SONGS. The team examined employee training, HP procedures, work planning, industry events, HP standards and jobsite work practices.

Several enhancements were identified by the assessment and will further reduce the station's likelihood of a PI related infraction. Below are some of the actions taken by the PI assessment team.

High Radiation Key Controls:

- management expectations were clarified in a new key control HP standard;
- capture key locks were installed for Tech Spec (>0.01 Sv)/(1R) hr areas;
- electronic HRA key inventories were developed vs. hard copy inventories;
- key tags identifying multiple HRAs controlled by one HRA key were instituted.

High Radiation Area Controls:

- shiftily walkdowns of Tech Spec HRAs (>0.01 Sv)/(<1 R/hr) areas were required;
- electronic HRA inventories were developed vs. hard copy inventory;
- operator instructions were clarified for multiple entries into HRAs.

Unintended Exposure Controls:

- revised work planning radiological risk assessments;
- included internal dose estimates for workers in TEDE ALARA evaluations;
- (i.e., in addition to collective dose estimates required by 10CFR20);
- included High Noise Areas evaluations in HRA job planning;
- routinely evaluated worker accumulated dose vs. anticipated dose
- developed performance standards and mock-up training for HP employees using SONGS remote radiation monitoring system;
- emphasised PI controls in HP contractor training.

The division also implemented a practice of identifying radiological PI precursor events and evaluating them for lessons learned. Figure 5 shows the San Onofre Nuclear Generating Station's Occupational Radiation Safety Cornerstone Performance Indicator profile for the past several quarters. Note the station has been in the upper band continuously, however this doesn't reflect precursor events or "close calls" which can inform management of degrading controls or human behaviors. Figure 5 shows precursor events which are associated with PI HP work practices. Although the pre-cursor events do not represent PI occurrences, they are helpful in assessing program controls. The table depicts off-normal events which the HPD has evaluated and taken additional action as warranted. The events on this table were generally attributed to HRA access control problems for areas $< (0.01 \text{ Sv})/1\text{R/hr}$ but $> (0.001 \text{ Sv})/100 \text{ mr/hr}$. The insights gained from evaluating these PI precursor events have benefited management's proactive decision making process.

Results

Several positive actions have been taken as a result of self-assessment activities. The radioactive material control program was strengthened by reconfiguring the containment equipment hatch controlled area during refuelings, implementing RCA tool control program changes and more stringent employee accountability measures. RCA access control measures were enhanced by installing an electronic dosimeter activated turnstile at the HP control points. Focused assessments drew the division's attention to improve the station's ALARA communications by conducting routine ALARA reports and ALARA briefings to better inform station management and employees of their dose minimization performance. Industry events are reviewed routinely for lessons learned during job planning and quarterly industry events training promotes employee discussion of the industry problems applicable at San Onofre Nuclear Generating Station.

The most recent WANO/INPO SONGS evaluation recognised the Health Physics Department self-assessment program for "effective assessment of Health Physics related activities to detect indications of declining performance." (INPO, August 2000).

Benefits of self-assessment

The benefits of a successful SA program are manifold and include the following:

- increased safety margin;
- increased regulatory confidence;
- improved performance;
- improved position in a competitive marketplace.

Regulatory confidence in a nuclear power plant licensee is essential. A successful self-assessment program will generate regulator confidence by implementing the elements of a program such as those described above. The U.S. NRC doesn't have the resources to perform assessments as the licensee nor do they have the station experiences or knowledge base of the licensee. Licensees who demonstrate sound self-assessment programs will likely receive credit from the U.S. NRC in the form of fewer inspections, favorable enforcement discretion, recognition of a healthy nuclear safety culture and the safe/consistent operations of the plant (Zimmerman, R.P., December 1998).

The U.S. NRC evaluates self-assessment programs using several Inspection Procedures. IP 40501 – Licensee Self-Assessments Related to Team Inspections calls for the U.S. NRC to examine the licensee’s capability to manage self-assessment and to conduct technically creditable self-assessments. IP 40500 – Effectiveness of Licensee Process to Identify, Resolve and Prevent Problems calls for the U.S. NRC to examine the licensee’s corrective action program, root cause analysis, self-assessments and operating experience feedback (Johnson, J.R., September 2000).

Self-Assessment is a good business practice in that it requires management to establish business expectations. These business expectations are captured as standards of conduct for which performance metrics are established and monitored. Self-assessment also requires management to be proactive by identifying and resolving precursors to prevent significant problems from occurring. In this manner self-assessment protects the company’s investment. Lastly, self-assessment programs improve employee morale and accountability through their participation/contribution in the self-assessment process, recognition of management’s responsiveness to employee observations and the positive results attributed to an active self-assessment program.

Future self-assessment challenges

The nuclear power industry will continue to change. Successful organisations will be led by continuously improving Self-Assessment programs. Self-Assessment Programs should focus on three key elements to promote continuous improvement (Krieger, R. W., September 2000).

The first is emphasising ongoing assessments by employees and management who continuously engage themselves in assessing human behavior and work process. The goal is to ensure employees are receptive to identifying and resolving problem areas; ensure management’s expectations are understood; routinely review and apply lessons learned from operating experiences; and implement a structured management/employee coaching and observation program.

The second is to further develop pre-emptive (forward-looking) indicators that use predictive rather than reactive approaches to assess performance. Typical station indicator or annunciator panels react to historical problems (rear view mirror approach) instead of anticipating or predicting problems.

The third is to routinely evaluate self-assessment program effectiveness and adjust program content as necessary. INPO has identified that reviews of self-assessment program effectiveness were weak. Better tools to evaluate program effectiveness and the value added need to be developed (INPO, December 1999).

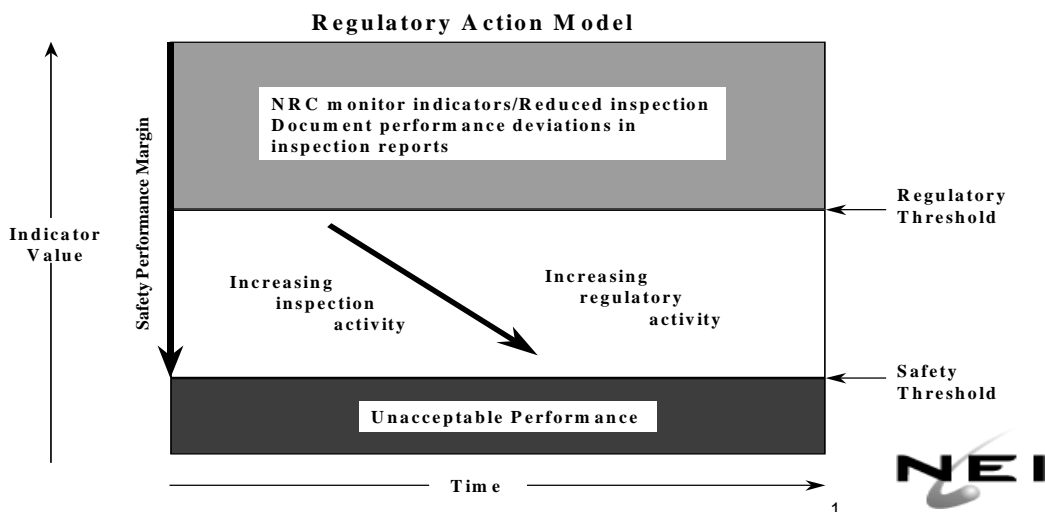
References

- [1] Institute of Nuclear Power Operations (INPO), WANO Peer Review, San Onofre Nuclear Generating Station, August 2000.
- [2] Institute of Nuclear Power (INPO), Principles for Effective Self-Assessment and Corrective Action Programs, December 1999.
- [3] Johnson, J.R., Associate Director for Inspections and Program, Office of Nuclear Reactor Regulation; Self-Assessments; ANS Workshop on Self-Assessment, September 2000.

- [4] Krieger, R.W., Vice President, San Onofre Nuclear Generating Station; The Value of Self-assessment; ANS Workshop on Self-assessment, September 2000.
- [5] Nuclear Energy Institute (NEI), Regulatory Assessment Performance Indicator Guideline, NEI 99-02, Rev. 0, March 2000.
- [6] San Onofre Nuclear Generating Station; Self-Assessment Program, Revision 1, November 1998.
- [7] Zimmerman, R.P., Deputy Director, Office of Nuclear Reactor Regulation; “Self-Assessment” Not a Program – A Way of Life; ANS Self-Assessment Executive Conference, December 1998.

Figure 1. NRC regulatory oversight framework

A NEW REGULATORY OVERSIGHT PROCESS
Risk-Informed, Performance Based Assessment, Inspection and Enforcement



CORNERSTONE CHART

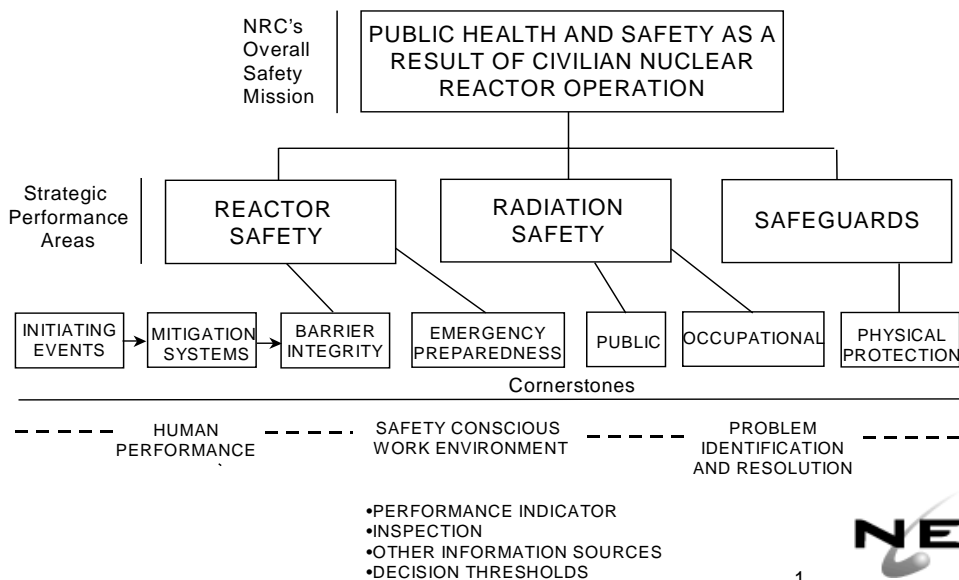
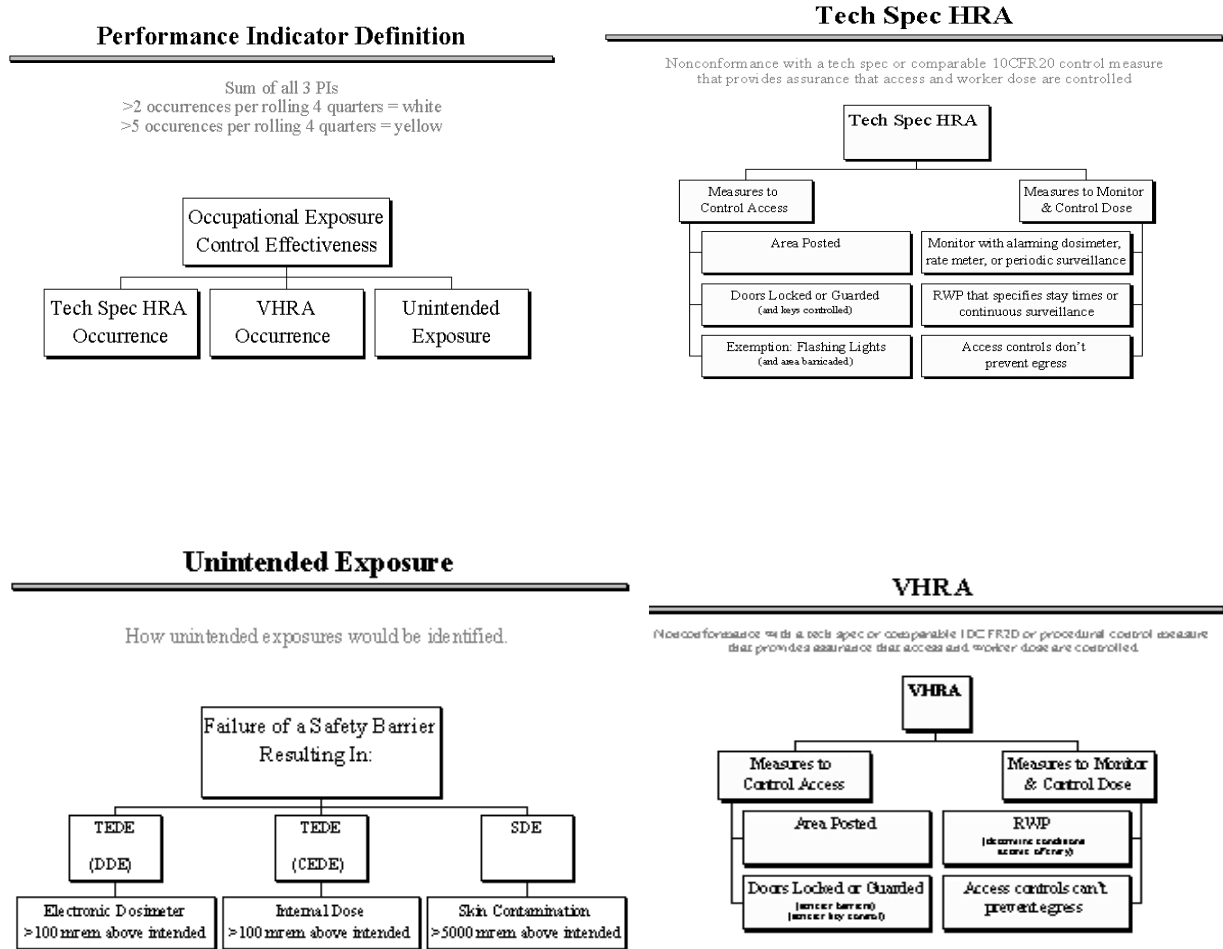


Figure 2. Occupational radiation safety cornerstones and performance indicators



Unintended Exposure

How unintended exposures would be identified.

Failure of a Safety Barrier Resulting In:

TEDE (DDE)

TEDE (CEDE)

SDE

Electronic Dosimeter >100 mrem above intended

Internal Dose >100 mrem above intended

Skin Contamination >5000 mrem above intended

VHRA

Nonconformance with a tech spec or comparable 10CFR20 or procedural control measure that provides assurance that access and worker dose are controlled.

VHRA

Measures to Control Access

Measures to Monitor & Control Dose

Area Posted

RWP (discipline conditions score of 0)

Doors Locked or Guarded (except barriers) (except key control)

Access controls can't prevent egress

Figure 3. SONGS integrated self-assessment program

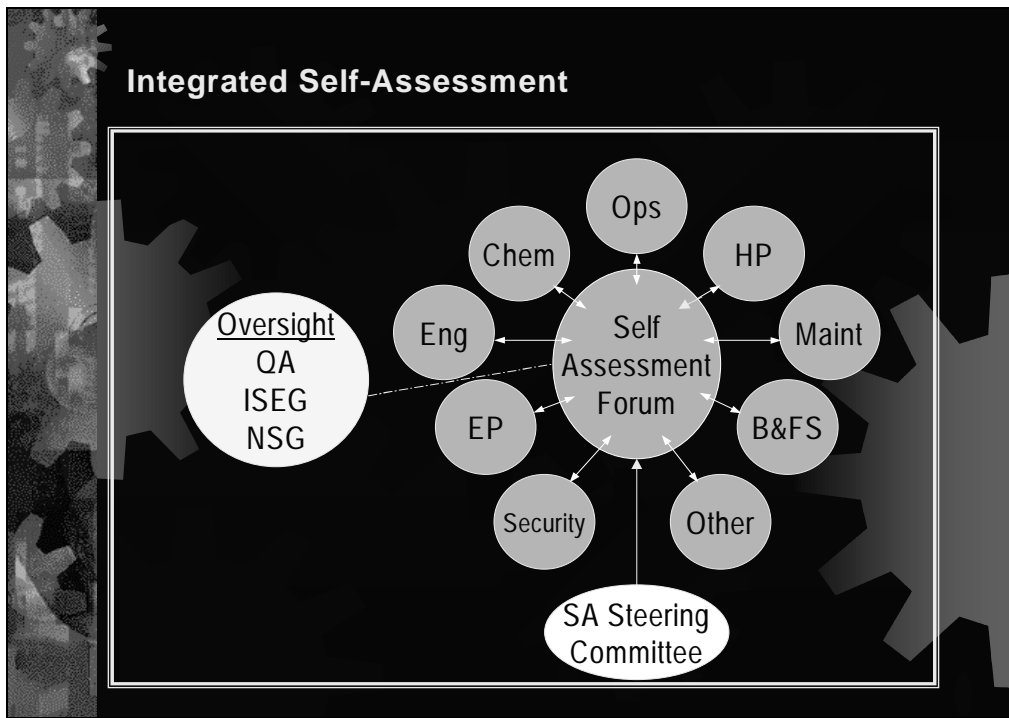


Figure 4. SONGS self-assessment program

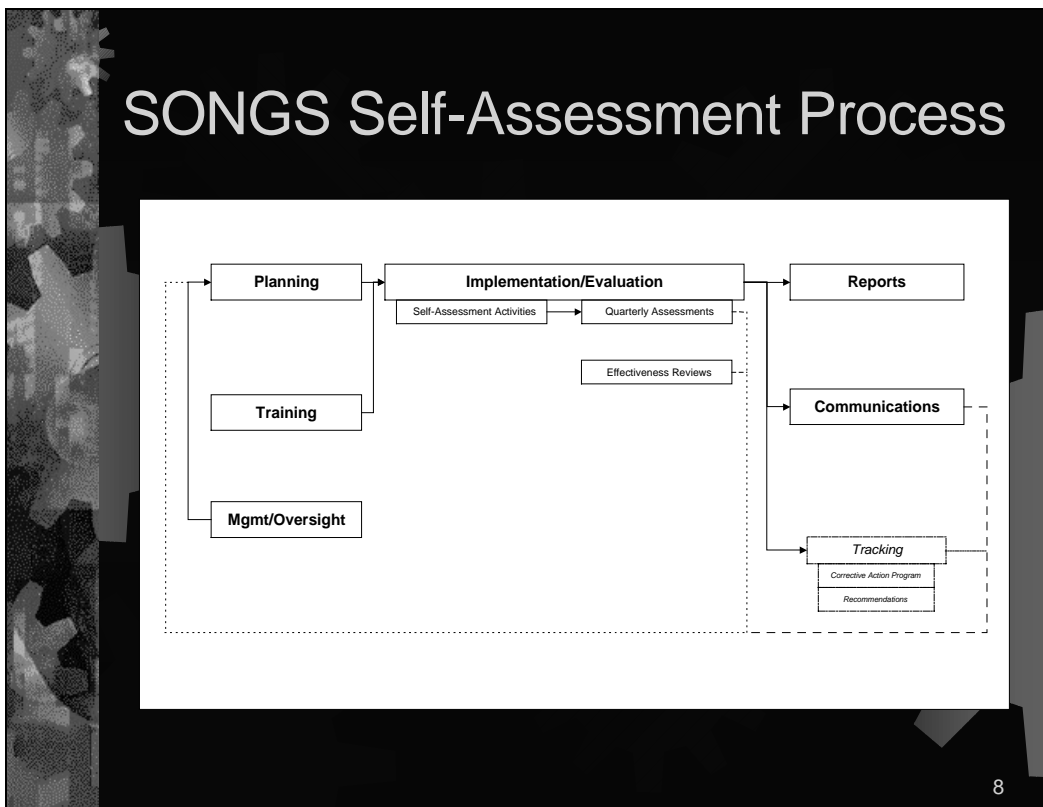
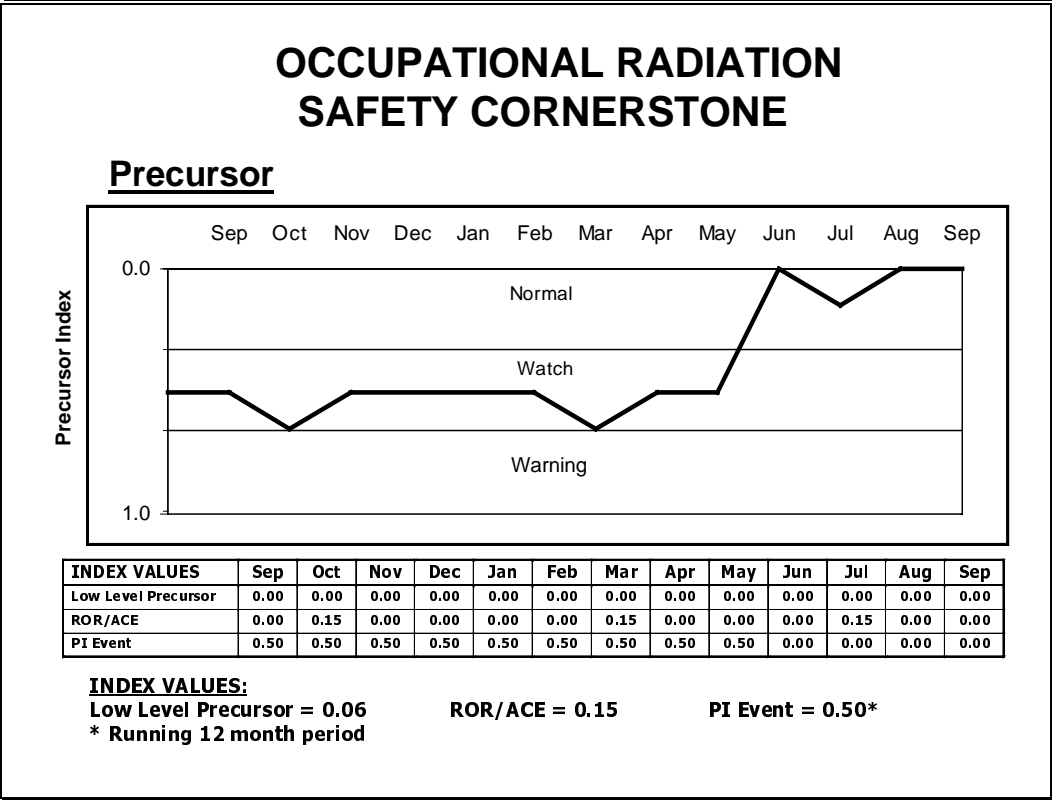
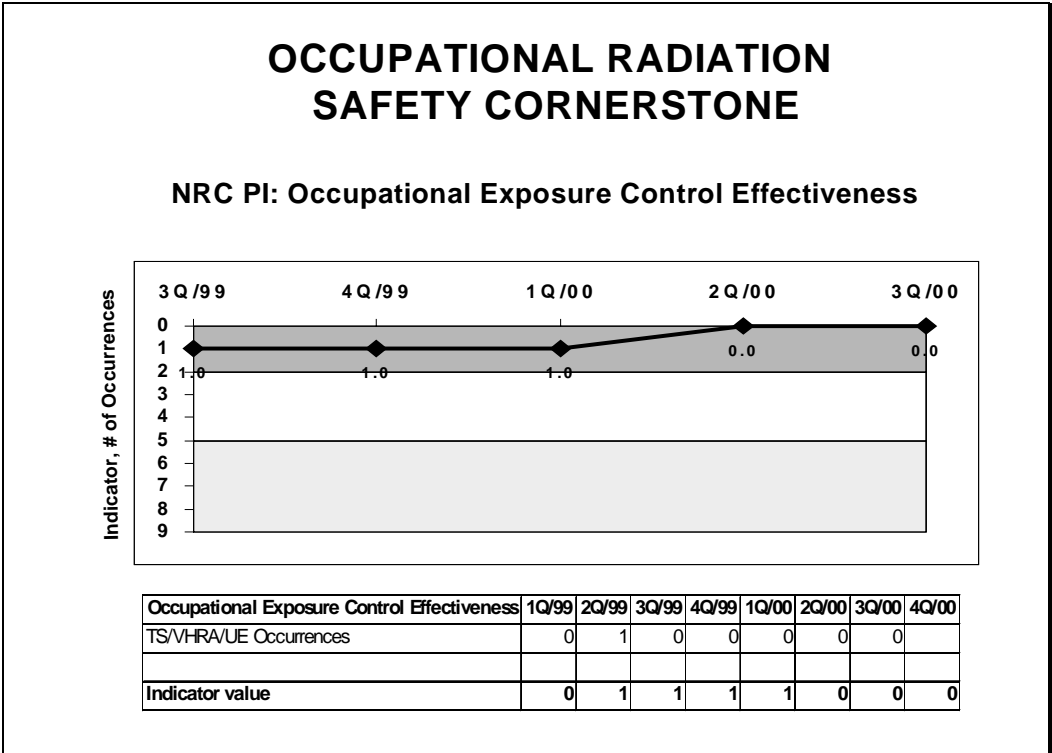


Figure 5. SONGS occupational radiation safety cornerstone and precursor results



SESSION 4

DISCRETE RADIOACTIVE PARTICLE ISSUES ASSOCIATED WITH A FUEL POOL CLEANOUT

R.L. Doty, R.C. Kessler
PPL SUSQUEHANNA, LLC, Canada

Abstract

As Susquehanna Steam Electric Station personnel were nearing the end of a project to remove control rod blades and other irradiated materials from the fuel pools at the station, the unit that was used to process the irradiated hardware (the “advanced crusher and shearer” or ACS) was removed from the cask storage pit. As the ACS was moved over the refueling floor to a location for further decontamination prior to shipment offsite, local area radiation monitors began to alarm. These alarms were due to one or more highly radioactive particles (up to 2.8 gigabecquerel) that were generated during the fuel pool cleanout (FPC) project and were inadvertently relocated from the pool to the surface of the refueling floor. Although these particles did not contact protective clothing or skin, these particles had the potential to deliver substantial doses to personnel in a very short period of time.

Over a period of several months, more discrete radioactive particles were found as a result of processing the irradiated hardware and performing other work evolutions during the FPC project. The highest estimated personnel doses were 0.12 and 0.17 Sv shallow dose equivalent, respectively.

The presentation will emphasize the lessons learned by Susquehanna personnel regarding planning for the FPC project, management oversight of the project, underlying cultural and mindset deficiencies that contributed to lack of preparedness for risks associated with highly active particles, and the steps for ensuring adequate contamination control.

Introduction

In developing the content for the 3rd EC/ISOE Workshop on Occupational Exposure Management at NPPs, the Program Committee proposed four topical sessions, including one on management of contamination control. The Committee suggested consideration of several aspects of contamination control-measurement, management, perception, and culture.

A series of contamination control events occurred in the last few months of the year 2000 at the Susquehanna Steam Electric Station, a two-unit boiling-water-reactor plant in the eastern United States of America. Those events challenged the personnel at the station and, as soon became clear, challenged some basic perceptions about the importance of contamination control as an element of exposure management. The intent of this paper is to briefly describe the situation and its industry-wide implications. By discussing those implications, a secondary result may be to leave the audience with a questioning attitude about the situation at their home plants. That is not to suggest weaknesses in any

plant's radiation safety program, but only to state that all radiation safety professionals can learn from the operating experiences at other stations.

Events of fall 2000

The Susquehanna station was conducting a clean-out campaign from the fuel pools in the latter half of 2000. The plan was to remove control rod blades, low power range monitors, and miscellaneous irradiated hardware from the pools, by processing and packaging the hardware for shipping and then transporting the materials to a licensed waste disposal facility. The project was to be of about five months in duration. About two months into the project, a key piece of equipment began to show degraded performance. The "advanced crusher and shearer" (ACS) was repaired and returned to service. Shortly thereafter, the forearm of a contract worker was exposed to a discrete radioactive particle, and a shallow dose equivalent of about 0.12 Sv was assigned. Refinement of radiation safety practices occurred over the next month as project activities continued.

When work with the ACS was completed, the ACS was removed from the cask storage pit and moved over the refueling floor to a laydown area for further decontamination prior to shipment off-site. During the movement of the ACS, a local plant-installed area radiation monitor began to alarm. On investigation, a 2.8 gigabecquerel (G bq) "hot particle", reading about 8 Sv/h on contact and consisting of cobalt-60, was discovered on the refueling floor. The likely source was displacement from the ACS as it was moved.

Recovery of the particle occurred over the next several days, with shielding and access control restrictions in place. A search for additional particles was conducted; three more contaminated areas were found. Additional radiation safety precautions were instituted, with some personnel surveyed for particles every 15 minutes.

Over the next two months, additional particles were identified. The two particles of highest activity were about 0.8 G bq and 0.7 G bq, respectively. No significant dose to personnel resulted from these particles. An additional personnel exposure occurred in early December 2000. An absorbed skin dose rate of 0.47 Gy/h resulted in an assigned shallow dose equivalent (SDE) of 0.17 Sv from a particle on an individual's protective shoe cover. Work on the project was suspended at this time. No doses in excess of any annual regulatory dose limits occurred during the project.

Near-term response to events

The station initiated several investigations during the course of the project. One root cause evaluation focused specifically on the event involving the discrete radioactive particle measuring 8 Sv/h. Another, later evaluation focused on the entire series of hot-particle events. Two evaluations focused on different aspects of project and refueling-floor management and independent oversight of the project. One evaluation even had as its emphasis a critique of the evaluation and investigation processes. Results will be described later in the paper.

After the events described above, the station's Radiation Protection Manager (RPM) left the company. The remaining investigative matters and the implementation of corrective actions were the responsibility of the incoming RPM.

The regulator (U.S. Nuclear Regulatory Commission or NRC) issued a Notice of Violation to the station for failure to conduct adequate evaluations and surveys. The emphasis of the NRC was

on the potential for doses exceeding the annual limit on Total Effective Dose Equivalent (TEDE), while it recognized that no workers accrued doses exceeding either the TEDE or SDE limits. That is, the NRC wished to bring attention to the potential for significant deep-dose equivalents given the high activities (and resultant high radiation fields) around some of the particles that were identified.

The detailed consideration of the “whole-body” dose implications of discrete radioactive particle exposures was an extension from the more common evaluations primarily for shallow (or skin) doses from exposures to hot particles. Listed in the references are some of the major documents relating to discrete radioactive particle exposures that were available by mid-2000 (1, 2, 9, 11). Neither the “deep” nor the “shallow” dose equivalent components of exposure may be overlooked for high-activity particles. In the case of a 2.8 Gbq particle of cobalt-60, a one-hour exposure can lead to doses of on the order of 2,900 Gy (skin, SDE) and 2.3 Gy (skin, DDE). Note should be made that means of calculating meaningful dose from exposure to discrete radioactive particles are under review. References 10 and 12 provide relevant information, and proposals for enhanced use of effective-dose-equivalent methodologies applicable to particle exposures are in review.

Lessons learned per regulator and industry organisations

The NRC, the Institute of Nuclear Power Operations (INPO), and the World Association of Nuclear Operators (WANO), have all released documents based on the events at Susquehanna. The NRC Information Notice (reference 13) brings forth the issue described above related to the potential for substantial dose to personnel, both for SDE and TEDE. The NRC describes also the need to consider adequacy of incorporation of previous plant and industry-wide experience into project planning, to ensure prevention of recurrence at Susquehanna or other stations.

INPO, in its Significant Event Report (reference 7), noted also the potential for significant unplanned exposures in short time periods. In addition, the INPO analysis listed the following contributors to the events:

- inadequate guidance for establishing a Hot Particle Control Zone (HPCZ);
- inadequate pre-job briefings;
- inadequate contamination control methods (e.g., inadequate hydrolysing and rinsing of the ACS prior to moving it); incomplete consideration of previous plant experience with discrete radioactive particles; and
- inadequate senior management presence and communication of high standards.

The staff at INPO are in the process of preparing a revision to its “Guidelines for Radiological Protection at Nuclear Power Stations” (reference 3). Increased discussion on discrete radioactive particles is to be included. A citation to the Susquehanna event (via reference 5) is included in the draft wording.

WANO, in its Significant Event Report (reference 14), stated also the potential for individuals to receive substantial unplanned exposures quickly. WANO then encouraged its member plants to consider the following:

- risk assessments that fully consider prior operating experience;
- contingency work plans for evolutions that may involve higher levels of risk; and
- adequacy of communications at pre-job briefings and during shift handovers.

Questions posed for appropriate discussion by radiation protection personnel and supervision are similar in the INPO and WANO documents. Those questions emphasize pre-planning of contamination controls and contingency planning for identification of highly radioactive particles.

Susquehanna perspective

The NRC, INPO and WANO documents described above are professionally done and should be useful to radiation protection staffs in their work planning. To supplement those evaluations, the items described below may also be useful to staff at other stations.

Root causes (RC) for the events at the Susquehanna station included the following five items:

1. Highly radioactive particles were generated during the fuel pool clean-out project and were removed from the pool and cask storage pit.
2. Personnel possessed an inaccurate risk perception of the dose consequence due to a discrete radioactive particle exposure. This was believed due to (a) the documented history of hot-particle exposure being a skin dose concern, and (b) research results identifying a lower risk from exposure to discrete radioactive particles [Workshop issue-perception].
3. Assessment processes that focused on actual personnel doses from particles that came in contact with either clothing or skin versus the potential doses that may have resulted from particles not found on personnel [Workshop issue – culture].
4. Inadequate pre-job plans with respect to control of particle exposures – survey techniques, instrumentation, and decontamination techniques were not sufficient [Workshop issues – management, measurement]; and
5. Previously identified cultural and performance issues that were not timely and effectively addressed – lack of a strong questioning attitude, inadequate communications, and incomplete use of operating experience had not yet been fully resolved [Workshop issue – culture].

Along with those causes were listed seven causal factors, which included items such as the inadequate procedural guidance and ineffective involvement of management that are stated in the analyses by the industry-wide organisations.

Putting the above issues into a different perspective, what is it that Susquehanna staff learned? First, there was a lack of sensitivity to the consequences of exposure to highly radioactive material (RC2) [Workshop issue – perception]. Associated with that was an attitude that significant radiological events couldn't happen at Susquehanna, a station with 18 years of operations. Also, the staff had to put more emphasis on the potential for exposure while at the same time addressing actual doses being accrued.

The staff learned that additional attention needed to be paid to contamination control planning (RC4), not only on the refueling floor but also in potentially impacted systems [Workshop issue – culture]. About a dozen systems received re-assessments of the potential for generation or release of highly active material. For the refueling floor, underwater vacuuming and filtration, hydrolazing, rinsing, and component covering were all reworked.

The need for structured, comprehensive pre-job briefings before work commences in a HPCZ is now fully recognized, as is the need for specific Radiation Work Permits (RWPs) for HPCZ work [Workshop issue – management]. In development of those RWPs, the use of respiratory protection to prevent the intake of particles needs to be carefully considered.

Radiation protection procedures have been revised to enhance radiological controls. Areas revised include survey techniques, labeling and posting, limiting dose rate and protective clothing requirements, and source term control. To supplement the enhanced survey techniques, instruments were modified for use in performance of surveys in HPCZs or when hot-particle presence is suspected. Radiation protection technicians received additional training on conducting particle surveys and on containment of identified particles [Workshop issue – measurement].

The self-assessment program is being revised to place additional emphasis on performance-based assessment of high-risk evolutions. Performance indicators have been developed. At the same time, procedures have been revised to more clearly identify high-risk evolutions and the enhanced precautions that must be taken for such jobs. Further, the need for additional management presence during the planning for and conduct of high-risk activities is considered on a case-by-case basis. Specific to contamination control, ongoing, critical self-assessment of practices is utilized and is being found to be effective in identifying and correcting contamination control deficiencies [Workshop issue – management].

Finally, there are two simply stated axioms: (1) be proactive to potential conditions rather than reactive to emergent conditions, and (2) when the mindset exists that “it can’t happen here”, it will.

Additional commentary

INPO, in reference 4, addresses precursors for unplanned exposures. Three of the root causes described above are aligned with the precursors described by INPO. That is, the adequacy of surveys and assessment (RC3), the adequacy of work planning and RWPs (RC4), and the effectiveness of supervisory direction and oversight (RC5) are correlated with listed precursors. For information, two other precursors noted by INPO were not directly applicable to this event; that is, neither incorrect guidance by radiological protection technicians nor non-compliance with radiological protection rules were contributors to the events at Susquehanna. The re-visiting of documents such as INPO’s SER 4-08 may be advisable for all plants as high-risk activities are planned.

The Susquehanna experience was certainly a focus of attention for numerous agencies for some time. Discrete radioactive particle events are not unique to Susquehanna, however. Examples of recent operating experience at other plants are described in references 6, 8 and 15. The point of mentioning those experiences is simple – particle events can happen at any plant. It is only with defense in depth through strong management, comprehensive planning, accurate perceptions of risk, a culture of awareness and risk control, and good measurement processes that prevention of recurrences will be successful.

References

- [1] Electric Power Research Institute, EPRI TR-104125, "Industry Experience with Discrete Radioactive Particles", July 1994.
- [2] Institute of Nuclear Power Operations, INPO 89-006, Good Practice RP-605, "Hot Particle Controls Program", July 1989.
- [3] Institute of Nuclear Power Operations, INPO 91-014 Revision 1, "Guidelines for Radiological Protection at Nuclear Power Stations", October 1995.
- [4] Institute of Nuclear Power Operations, SER 4-98, "Unplanned Personnel Radiation Dose", 23 November 1998.
- [5] Institute of Nuclear Power Operations, Operating Experience Report No. OE11779, "Highly Radioactive Particles Associated with Spent Fuel Pool Work", 27 December 2000.
- [6] Institute of Nuclear Power Operations, Operating Experience Report No. OE12059, "Hot Particle Incident", 28 March 2001.
- [7] Institute of Nuclear Power Operations, SER 3-01, "Highly Radioactive Particles Associated with Fuel Pool Work", 30 March 2001.
- [8] Institute of Nuclear Power Operations, Operating Experience Report No. OE13293, "Inadequate Documentation of Hot Particle Event", 15 February 2002.
- [9] National Council on Radiation Protection and Measurements, NCRP Report No. 130, "Biological Effects and Exposure Limits for Hot Particles", 10 December 1999.
- [10] National Council on Radiation Protection and Measurements, NCRP Statement No. 9, "Extension of the Skin Dose Limit for Hot Particles to Other External Sources of Skin Irradiation", 30 March 2001.
- [11] U.S. Nuclear Regulatory Commission, Information Notice 90-48, "Enforcement Policy for Hot Particle Exposures", 2 August 1990.
- [12] U. S. Nuclear Regulatory Commission, Proposed Rules, 10CFR20, "Revision of the Skin Dose Limit", Federal Register 66, No. 134, 36502-36509, 12 July 2001.
- [13] U.S. Nuclear Regulatory Commission, Information Notice 2002-03, "Highly Radioactive Particle Control Problems During Spent Fuel Pool Cleanout", 10 January 2002.
- [14] World Association of Nuclear Operators, WANO SER 2001-2, "Highly Radioactive Particles Associated with Fuel Pool Work", August 2001.
- [15] World Association of Nuclear Operators, WOE00430, "PHWR Event – High Extremity Dose Received from Hot Particle (9.6 rem)", 21 December 2001.

CONTROLLING THE ALPHA RISK IN THE EDF FACILITIES

A. Roupioz

EDF CAPE/GPR, France

J.F. Labouglie

CNPE CATTENOM, France

Foreword

Due to the characteristics of the radiation they emit, alpha emitters are highly radiotoxic in terms of internal exposure but their low penetration makes them inoffensive in terms of external exposure. This toxicity is clearly shown in the table below, which compares the equivalent dose resulting from the ingestion of 1 Bq or 1 μ g (10^{-6} g) of a few radionuclides commonly found in nuclear power plants:

Radionuclide	Type of emission	Equivalent dose due to the ingestion of 1 Bq	Dose equivalent to the ingestion of 1μg
Plutonium 239	α	140 μ Sv	0.32 Sv/ μ g
Curium 244.	α	76 μ Sv	21.71 Sv/ μ g
Americium 241	α	140 μ Sv	17.5 Sv/ μ g
Cobalt 60	β + γ	0.06 μ Sv	2.46 Sv/ μ g
Cesium 134	β + γ	0.013 μ Sv	0.54 Sv/ μ g
Iodine 131	β	0.008 μ Sv	33.75 Sv/ μ g
Tritium	β	0.000016 μ Sv	0.01 Sv/ μ g

In PWR power plants, α emitters are rare and the risk of contamination is particularly low. Until now, this risk has always been well controlled by the operator, and is not one of the main Radiological protection priorities within EDF.

Nevertheless, requirements are always increasing, whether for the radiological protection of workers or the public, and have led EDF to reconsider the α risk, to ensure that the current provisions are adequate and to reinforce them if necessary.

Therefore, a study of the α risk was started in 1999.

The production of α emitters in PWRS

The existence of alpha emitters in the primary cooling system has various origins:

- The main source of contamination is associated with the damage of fuel during operation. If the faults are serious enough, physical damage and erosion of uranium oxide in contact with the primary coolant may occur.
- To a lesser extent, alpha emitters are due to the activation of the small quantities of fissile material on the fuel rods before loading. The presence of such fissile material is associated with residual contamination of cladding in the factory, or with the presence of natural Uranium in the cladding material.

These fine particles in suspension in the primary coolant are deposited fairly quickly, mainly in heat exchange areas and on the surface of fuel rod cladding, but also in dead legs, filters and walls.

The particles deposited on the walls of the primary cooling system are diffused within the layers of oxides and are fixed on them. Under such conditions, only part of the actinides is retained by the purification circuit, the other part remains in the components of the primary cooling system and is likely to be released during maintenance work, long after the initial pollution.

We must therefore distinguish between two types of contamination of the primary cooling system:

- that of the fluid itself, which is monitored by the radio-chemical monitoring of the water of the primary cooling system:
 - the presence of iodine 134 is a good indicator of the presence of fissile material;
 - the α contamination of the primary cooling system which is measured periodically.
- that of the walls of the primary cooling system out of flow¹, which builds up during the life of the power plant, by “the memory effect” of the layers of oxides, and which therefore depends as much on the history of the unit and the faults which occurred during previous cycles, as the state at of the fuel cladding in the current cycle.

There is therefore a risk of internal exposure of workers working on components of the installation.

Prevention of the α risk for workers

The real goal for EDF is to not create internal α contamination of workers.

In view of that described above, the risk of contamination is mainly due to maintenance and dismantling work, when, during such work, the layers of oxides in the circuits in contact with the primary coolant are subject to mechanical attack (by grinding, cutting, etc...) or when the surface contamination is resuspended by the ambient air scavenging of non fixed particles.

1. “In flow” contamination is eliminated during the replacement of fuel.

Such dust may be:

- ingested or inhaled directly by workers if they are not equipped with personal protection (helmet or ventilated clothing, etc.);
- or deposited on others parts of the installation if the work site is not confined, thus extending the area at risk of contamination.

Not all facilities have the same level of potential alpha risk. This risk is directly associate with the operating history of the installation and the nature of the work to be carried out.

We estimate that for PWRs, the average level of surface contamination of circuits in alpha emitter radionuclides is between 0.5 and 1 Bq/cm².

Type of monitoring adopted by edf

EDF has decided to acquire a systematic installation monitoring system with three levels of investigation:

1. monitoring of the primary cooling water and pools in operation, as an initial alert indicator and to anticipate the special provisions to be taken;
2. monitoring at the start of a unit shut-down, of certain highly exposed surfaces of the installation particularly to determine whether an α risk is to be taken into consideration more extensively;
3. if the second level is positive, monitoring the possible α contamination of work site likely to be contaminated.

For the first level of monitoring, an Iodine 134 activity threshold and an alpha activity measurement of the primary coolant or of the spent fuel pit enables the risk of α contamination to be anticipated before the next shut-down. The shut-down is therefore prepared by taking the special provisions into account and we may revise such provisions if the contamination is not finally confirmed.

The second level of monitoring consists of radiological protection measures against labile alpha contamination (unstable) on surfaces not directly in contact with fuel elements but in contact with the primary coolant, such as the pressuriser and the reactor vessel head.

Such surfaces correspond to sensitive areas, which represent the maximum contamination of the installation. This level of monitoring is also used to take the history of the installation into account.

An installation is declared to be “at alpha risk” when the radiological protection measures reveal labile alpha contamination greater than 8 Bq/cm². This value corresponds to the resuspension of labile surface contamination of Am 241, leading to atmospheric contamination of 1 LDCA when the resuspension factor is equal to 10⁻⁶ m⁻¹.

Where second level monitoring is positive (>8 Bq/cm²), that is the installation is considered to be “at alpha risk”, third level monitoring is started and consists in checking for the presence (or not) of alpha contamination on all parts of the installation contaminated by activation products (rooms, equipment, ventilation, etc.).

Under such conditions, routine prevention provisions against internal exposure are taken and personal and collective protection equipment are used.

Special attention is paid to grinding work which requires the systematic use of prevention devices. Therefore, for installations being dismantled and maintenance workshop, the activities of which include grinding work, the provisions taken for the confinement and protection of workers are applied systematically.

Implementation and information feedback

The principles and provisions described above were implemented in the Cattenom plant, in particular during the shut-down of unit No. 3 in 2001 and were found to be totally satisfactory.

DOSE ASSESSMENTS FROM WHOLE-BODY MEASUREMENTS – THE APPROACH AT NPP KOZLODUY

G. Valtchev, L. Dimitrov, M. Neshkova
NPP Kozloduy, Bulgaria

Abstract

Dose assessments from intakes of inhaled radionuclide present rather complex problems. It requires some information and comprehensive investigation and calculation. In addition ICRP has reviewed and changed some of the biokinetic models and adopted a new model for the human respiratory tract (ICRP Publications 56, 66, 67, 69). At NPP Kozloduy two programmes were applied for such type of dose assessments: CINDY 1.3 (USA), and AIDA (Activity and Intake Dose Assessments) (Local one), both based on ICRP Publication 30. The above circumstances were reflected in a new software named DOSEART, designed to meet the Bulgarian Radiation Safety Rules (ONRZ 2000) as well.

The program DOSEART is verified against LUDEP 2.07 for some typical for NPP radionuclides. It is developed using Visual Basic 5.0 and is quite user-friendly and reliable.

Introduction

The programme for monitoring the internal contamination at Nuclear Power Plant Kozloduy (KNPP) is based entirely on *in vivo* whole body measurements as the most convenient and rapid method for determining uptakes of radioactivity by inhalation. The counting system is Canberra type: Accuscan scanning bed position and the appropriate software for the estimation.

Whole-body counting determines the amount of radioactivity present in the body at the time of the measurements (an uptake), but it cannot determine the amount present at some previous time; that quantity must be inferred from the measured body content and from the application of metabolic models or retention curves that describe the behavior of the radionuclides in the body. Routine individual monitoring consists of regularly repeated measurements made on individual worker, while special monitoring is performed before and after particular task. The estimation of doses from radionuclides within the body (internal emitters) is less straightforward than with external radiation – internal dose cannot be measured directly. In the interpretation of the measurements the major source of uncertainty lies in the unknown intake times.

Internal dose assessment

The KNPP's programme is based on routine and partly on special individual monitoring. Thousands of measurements are performed annually in the whole-body laboratory. If any activity is registered then the Health Physicist (HP) has to answer a couple of important questions:

- Is it external or internal contamination?
- What is the chemical form of the nuclides?
- When did it happen?
- What is the size of the inhaled aerosol (AMAD)?
- Did all measured nuclides have the same size (AMAD)?
- Do we have one or more intakes between the actual and previous measurements.
- How to estimate the actual if a previous intakes were registered.

On the other hand to estimate the dose from the measured activity we should know the biokinetic behavior of the particular radionuclide:

- The accumulation of activity in specific organs.
- The retention of radionuclides in those organs.
- The transport of radionuclides between organs.
- The removal by excretion and radioactive decay, of activity from the body.

To answer all these questions the HP should be quite experienced. The following two phases are inevitable when estimating the internal dose:

- Questioning phase.
- Investigation phase (individual metabolism).

The very important instrument, needed by the HP, is an actual user-friendly software. This paper will not deal with the mentioned above phases. We want to present our software that helps to calculate rapidly different input data and compare the results for those inputs.

The software "DOSEART"

At NPP Kozloduy two programs were applied for such type of dose assessments: CINDY 1.3 [1] (USA), and AIDA (Activity and Intake Dose Assessments), local one, both based on ICRP Publication 30. Recently ICRP has reviewed and changed some of the biokinetic models and adopted a new model for the human respiratory tract (ICRP Publication 66 [2]). The above circumstances were reflected in a new software named DOSEART, designed to meet the Bulgarian Radiation Safety Rules (ONRZ 2000) as well.

The program DOSEART is verified against LUDEP 2.07 [3] for some typical for NPP radionuclides. It is developed using Visual Basic 5.0 and is quite user-friendly and reliable. To fully describe the software will take more time and pages. So only some main features will be discussed here.

The values of the whole body intake retention fractions (IRF) from inhalation have been calculated in advance by preliminary developed application programmes – IRF generators. Whole body's compartment model and matrix notation were applied. The IRF values are saved in form of a library, (data files), for different radionuclides, for AMADs 1 and 5 microns, types F, M, S, and time after intake (up to 700 days in step of 1 day). One IRF generator is based on the models as in ICRP Publication 30. The second one was created using the ICRP Publication 66 introducing the new lung model. Below the table gives values of IRF for different types and radionuclides, calculated with the second IRF generator and LUDEP 2.07.

Days	Type S, Co – 60 AMAD 1 micron		Type F, CS – 137 AMAD 1 micron		Type S, Co – 60 AMAD 5 micron		Type F, CS – 137 AMAD 5 micron	
	Inhalation IRF	LUDEP 2.07	Inhalation IRF	LUDEP	Inhalation IRF	LUDEP	Inhalation IRF	LUDEP 2.07
1	0.3447	0.3469	0.4881	0.4922	0.4881	0.4922	0.5974	0.5944
2	0.2202	0.2219	0.2508	0.2540	0.2508	0.2540	0.5062	0.5020
3	0.1617	0.1625	0.1417	0.1430	0.1417	0.1431	0.4663	0.4622
4	0.1373	0.1376	0.0974	0.0978	0.0974	0.0978	0.4466	0.4432
5	0.1272	0.1272	0.0798	0.0799	0.0798	0.0799	0.4352	0.4325
6	0.1225	0.1225	0.0726	0.0725	0.0726	0.0725	0.4279	0.4255
7	0.1201	0.1201	0.0693	0.0692	0.0693	0.0692	0.4223	0.4203
8	0.1184	0.1184	0.0675	0.0674	0.0675	0.0674	0.4178	0.4160
9	0.1171	0.1170	0.0663	0.0662	0.0663	0.0662	0.4138	0.4123
10	0.1158	0.1158	0.0653	0.0652	0.0653	0.0652	0.4101	0.4090
20	0.1061	0.1060	0.0584	0.0583	0.0584	0.0583	0.3828	0.3822
30	0.0985	0.0985	0.0536	0.0536	0.0536	0.0536	0.3593	0.3585
40	0.0926	0.0926	0.0500	0.0499	0.0500	0.0499	0.3366	0.03365
50	0.0877	0.0876	0.0471	0.0471	0.0471	0.0471	0.3160	0.3157
60	0.0836	0.0836	0.0447	0.0447	0.0447	0.0447	0.2967	0.2963
70	0.0801	0.0802	0.0427	0.0427	0.0427	0.0427	0.2796	0.2780
80	0.0774	0.0773	0.0412	0.0411	0.0412	0.0411	0.2605	0.2608
90	0.0746	0.0747	0.0396	0.0397	0.0396	0.0397	0.2445	0.2447
100	0.0726	0.0726	0.0385	0.0385	0.0385	0.0385	0.2296	0.2296

In case of a multiple or a continuous intake, the total intake I_{TOTAL} could be approximated by an effective single intake and its relation to the whole body activity $A(T)$, is given by the following expression:

$$I_{TOTAL} = \frac{A(T)}{IRF_{EFFECTIVE}}$$

In case of continuous intake for a period T, the expression for the effective IRF is as follows:

$$IRF_{\text{EFFECTIVE}} = \frac{A(T)}{I_{\text{TOTAL}}} = \frac{\int_0^T I(t) \cdot IRF(T-t) dt}{\int_0^T I(t) dt}$$

In case of multiple N intakes, the expression is:

$$IRF_{\text{EFFECTIVE}} = \frac{A(T)}{I_{\text{TOTAL}}} = \frac{\sum_{k=1}^n I_k \cdot IRF_k}{\sum_{k=1}^n I_k}$$

Using the assumption for a constant intake rate, the above expression is simplified to:

$$IRF_{\text{EFFECTIVE}} = \frac{1}{T} \int_0^T IRF(t) dt$$

$$IRF_{\text{EFFECTIVE}} = \frac{\sum_{k=1}^n IRF_k}{n}$$

To calculate the dose from the estimated intake, the appropriate dose coefficients from BSS 115 [4] are used. They are tabulated in the programme for the selected types, AMAD and radionuclides. The advantages of the software “DOSEART”, when comparing with older versions as Cindy, should be quite easy understandable by experienced in the topic Health Physicist. Some of them are listed below:

- the time for input data and calculations is much shorter;
- easy to see and change all input data;
- previous intakes are taken into account when estimating the actual one;
- possibility to estimate multiple intakes;
- calculation modes for special and routine monitoring;
- calculated intakes and doses are given at the same time;
- when input of many radionuclides, all intakes and doses are partially presented;
- percentages of ALI for each of the radionuclides are presented;
- the programme automatically changes the values when dimensions are changed;
- doses for continuous intake are evaluated;
- implementation of the new lung model.

No report output is considered disadvantage of the software. However DOSEART is used in the investigation phase and for theoretical studies. When necessary the values are put into the legal certificate form.

Four typical screens of the programme are shown at the end. They illustrate different modes and calculation for selected radionuclides.

References

- [1] CINDY, Batelle Pacific Northwest Laboratories, USA, DOE Order 5480.11 , 1992.
- [2] ICRP Publication 66, Human Tract Model for Radiological Protection, Ann. ICRP 24, 4. Pergamon Press, Oxford UK, 1994.
- [3] Jarvis N.S., Birchall A., James A.A., Bayley M. R. and Dorian M.D , LUDEP 2.0 -Personal computer programme for calculating internal doses using the ICRP Publication 66 respiratory tract model, Chilton, NRPB -SR 287 , 1996.
- [4] International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No.115, IAEA, 1996.

DOSE ART New Lung Model (ICRP publication 66)

Radionuclide DATA Additional DATA Final RESULTS

Measurement Time [seconds] 300

ACTIVITY INTAKE DOSE
microCi Bq mSv

Personal Data
Age (years) []
Weight (kg) 70

Monitoring
 Routine
 Special

Routine Monitoring
Measurement: [dd.mm.yy]
current 10.04.02
previous 18.01.02
before previous 13.10.01

Intake
 Acute
 Multiple
 Continuous

24.11.01 02.03.02

Nuclide	Class	Amad	Bq	Bq	microCi	INTAKE	DOSE
Co-58							
Co-60	S	5	1000	2000		0.027	18494 0.31
Cs-134							
Cs-137	F	5	1000			0.027	2945 0.02
Mn-54	M	5	1000			0.027	16681 0.02
Ag-110m	S	5	1000			0.027	18871 0.14
Zr-95							
Nb-95							
Cr-51							
Fe-59							
Zn-65							
Ru-103							
Ru-106							
I-131							
Ce-141							
Ce-144							
TOTAL:							56381 0.49

Calculate New Data End

Company: [] Name: []
Department: [] Code: [] Print
Sector: [] Comments: []

DOSE ART New Lung Model (ICRP publication 66)

Radionuclide DATA Additional DATA Final RESULTS

Measurement Time [seconds] 300

ACTIVITY INTAKE DOSE
microCi Bq mSv

Personal Data
Age (years) []
Weight (kg) 70

Monitoring
 Routine
 Special

Routine Monitoring
Measurement: [dd.mm.yy]
current 10.04.02
previous 18.01.02
before previous 13.10.01

Intake
 Acute
 Multiple
 Continuous

24.11.01 02.03.02

Nuclide	Class	Amad	Bq	Bq	microCi	INTAKE	DOSE
Co-58							
Co-60	S	5	1000	2000		0.027	18494 0.31
Cs-134							
Cs-137	F	5	1000			0.027	0.10 0.02
Mn-54	M	5	1000			0.027	0.10 0.02
Ag-110m	S	5	1000			0.027	0.69 0.14
Zr-95							
Nb-95							
Cr-51							
Fe-59							
Zn-65							
Ru-103							
Ru-106							
I-131							
Ce-141							
Ce-144							
TOTAL:							2.46 0.49

Calculate New Data End

Company: [] Name: []
Department: [] Code: [] Print
Sector: [] Comments: []

DOSE ART New Lung Model (ICRP publication 66)

Radionuclide DATA Additional DATA Final RESULTS

Measurement Time [seconds] 300

ACTIVITY INTAKE DOSE
microCi Bq mSv

Personal Data
Age (years) []
Weight (kg) 70

Monitoring
 Routine
 Special

Routine Monitoring
Measurement: [dd.mm.yy]
current 10.04.02
previous 18.01.02
before previous 13.10.01

Intake
 Acute
 Multiple
 Continuous

24.11.01 02.03.02

Nuclide	Class	Amad	Bq	Bq	microCi	INTAKE	DOSE
Co-58							
Co-60	S	5	1000	2000		0.027	13150 0.22
Cs-134							
Cs-137	F	5	1000			0.027	2839 0.02
Mn-54	M	5	1000			0.027	12300 0.01
Ag-110m	S	5	1000			0.027	13634 0.10
Zr-95							
Nb-95							
Cr-51							
Fe-59							
Zn-65							
Ru-103							
Ru-106							
I-131							
Ce-141							
Ce-144							
TOTAL:							41983 0.36

Calculate New Data End

Company: [] Name: []
Department: [] Code: [] Print
Sector: [] Comments: []

DOSE ART New Lung Model (ICRP publication 66)

Radionuclide DATA Additional DATA Final RESULTS

Measurement Time [seconds] 300

ACTIVITY INTAKE DOSE
microCi Bq mSv

Personal Data
Age (years) []
Weight (kg) 70

Monitoring
 Routine
 Special

Routine Monitoring
Measurement: [dd.mm.yy]
current 10.04.02
previous 18.01.02
before previous 13.10.01

Intake
 Acute
 Multiple
 Continuous

24.11.01 02.03.02

Nuclide	Class	Amad	Bq	Bq	microCi	INTAKE	DOSE
Co-58							
Co-60	S	5	1000	2000		0.027	9145 0.24
Cs-134							
Cs-137	F	5	1000			0.027	2839 0.02
Mn-54	M	5	1000			0.027	12300 0.01
Ag-110m	S	5	1000			0.027	13634 0.10
Zr-95							
Nb-95							
Cr-51							
Fe-59							
Zn-65							
Ru-103							
Ru-106							
I-131							
Ce-141							
Ce-144							
TOTAL:							36378 0.37

Calculate New Data End

Company: [] Name: []
Department: [] Code: [] Print
Sector: [] Comments: []

CONTAMINATION CONTROL AT THE EXIT FROM RADIATION CONTROLLED AREA AND NPP SITE AT BOHUNICE NPP

S. Jaroslav
Bohunice NPP, Slovak Republic

Introduction

Bohunice Nuclear Power Plant is situated in south – western part of Slovakia about 50 km away from Bratislava. There are four PWR reactors 440 MWe each – two units with reactors VVER-230 (V1 NPP) and two units with VVER-213 (V2 NPP) (Table 1).

Table 1. **Bohunice NPP – List of reactor units**

Name	Unit	Operation
V1	PWR, Unit 1, VVER-230	1978
	PWR, Unit 2, VVER-230	1980
V2	PWR, Unit 3, VVER-213	1984
	PWR, Unit 4, VVER-213	1985

The original design of four reactor units at Bohunice NPP came from seventies years of 20th century. The same was valid for radiation protection instrumentation.

The process of safety improvement and operational reliability started immediately after the units commissioning. Based on the result of internal experience and international review missions (OSART, WANO, ...) a large attention was put in contamination control at the radiation control areas boundaries, personal and vehicle gate site.

Contamination monitors at RCA boundary

The layout of hot and cold change rooms was rebuilt in order to enhance their hygienic status and to unify the previous multiple entry/exit points into only one controlled point. There are two different contamination monitors at the exit from RCA. The measuring of surface contamination of clothes is performed by original Russian contamination monitors (RZB-04-04, RUSSI-1) at the first monitoring point. The second monitoring point includes the contamination monitors for personnel dressed only in underwear (APM3A) and the activity monitors (CPO) for the small objects taking out of radiation controlled area. The modernisation of the contamination monitors was made only for contamination control of personnel.

The modernization of the contamination monitors at the exit from radiation controlled areas began in 1998. The innovated system of contamination measurement at RCA exit was put in operation in May 2001.

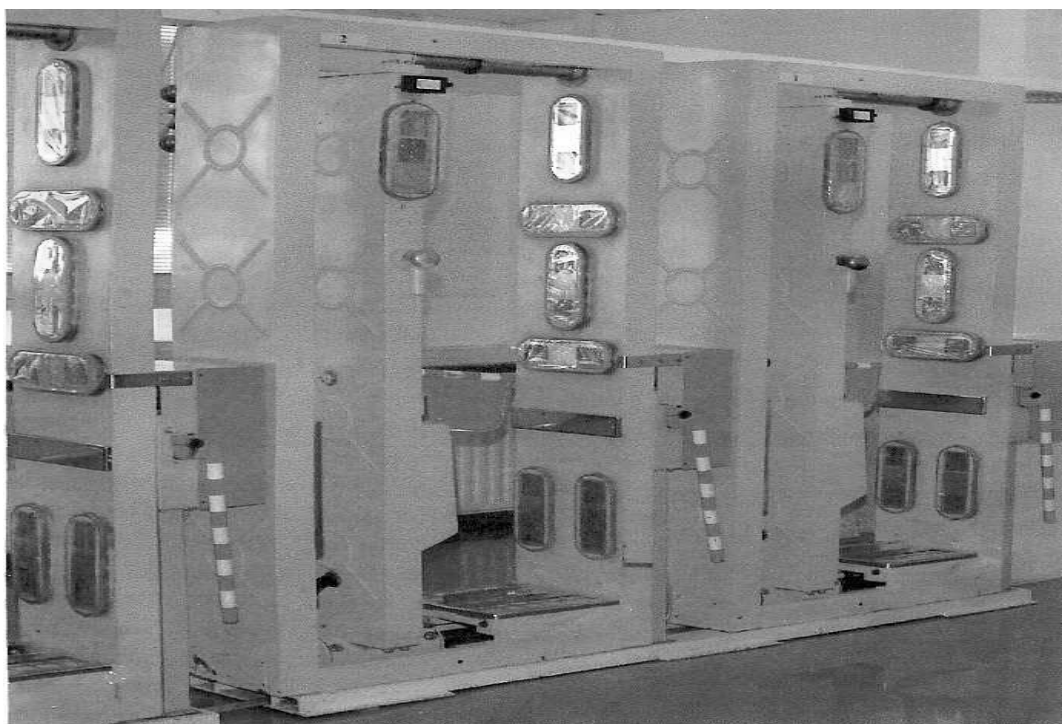
Description of monitors

RUSSI-1, RZB-04-04 monitors

The old RUSSI-1 and RZB-04-04 monitor uses fifteen detectors split into front and back panel, each detector contains two GM tubes type SBT-10. Contamination alarm levels can be set for individual detector. There are entry/exit barriers, the measuring time is more than 5 second and the active monitoring area is 1 071 cm². The detection surface and the dead spacing between detectors are main disadvantages of those monitors.

At present the old beta gamma monitors are used for contamination control of individuals wearing overalls at the entrance to the hot changing room. The calibration source is ⁹⁰Sr and 3 Bq/cm² is the alarm level set for each individual detector.

Figure 1. **RUSSI-1 monitors**



APM 3A monitor

APM-3A is two steps alfabeta and gamma-ray sensitive whole-body monitor. There are large gas flow proportional detectors split into some internal compartments for monitoring the hands, body, feet, head and small objects. The monitor comprises twenty four large area LFP-800 detectors and two

LFP-330 detectors. The active monitoring area is 32 000 cm² for the body and 3 200 cm² for small objects. There are entry/exit barriers, foot, hands and body photosenzors and reader for ID cards. The control takes place in two steps: body front and one side first, then back and other side. The measurement time is more than eight second for one step.

The APM-3A monitors are used for monitoring of the surface contamination of personnel at the exit from the hot changing-room. The alarm set point is 45 Bq for each detector as well as for sum-zone. Radioisotope ¹³⁷Cs is used for calibration. The APM-3A is highly-sensitive monitor with an excellent coverage of body surface.

CPO monitor

The CPO uses two large area scintillation detectors (1 225 cm² each) in the top and bottom sites. There also is the lead shielding (25 mm), entry/exit barriers and audible warning.

The CPO monitors measure the small objects taking out of radiation controlled area. The calibration is performed by ⁶⁰Co and the alarm level is 90 Bq. The complex of APM-3A and CPO monitors creates the radiological contamination barrier at the exit from the hot changing-room which is the RCA boundary.

Figure 1. CPO and APM-3A monitors



Activity monitor at NPP site (personal gate)

Several instruments PM7 had been integrated into the overall security system of NPP. The condition for integration was to do not restrict the number of passing people. The PM7 monitor is equipped with the plastic scintillation detectors and with the lead shielding. Scintillators are distributed partly in each pillar (left, right) and in top and bottom part of the monitor. The RDA (Reliable Detectable Activity) of PM7 monitors lies within the interval (9.2-10.4) kBq for ^{137}Cs (dotted source in the middle of the monitor).

The innovated system of contamination measurement at the personal and vehicle site gates was put in operation in January 1998.

Figure 2. PM7 monitors



Experience with installation

PM 7

The measuring algorithm of PM7 had been accommodated during the installation in order to facilitate fluent passing of the personnel through the monitors.

APM-3A

There were larger problems with the control of the P10 gas flow of APM-3A monitors because the gas supply system is situated outside the building.

Immediately after the putting the monitors into the operation a large number of shoe soles contamination had been indicated by those monitors. Those shoes are used only for passage from the hot to the cold room. The main cause of the contamination was the superposition of the small contamination existing on the floor of the hot change room on the shoes' soles. The accepted corrective actions were: more frequent washing of the hot change rooms floors and positioning of the deco foils at the entrance to the hot change rooms

Site gate contamination

The events of the contamination at the site gate of the NPP are logged from 1998 when the personnel and vehicle contamination monitors were put in operation. In this paper there are only events revealed by PM 7 personal monitors reported.

The contamination review is divided into two periods.

The first period from 1st January 1998 to 1st June 2001 is characterised by the fact that the modernisation of the contamination monitoring equipment at the exit from RCA had not been finished yet and thus still the old monitors RUSI-1 and RZB 04-04 had been in operation.

In the second period (1st June 2001 – 28th February 2002) the old monitors were replaced by new ones APM 3A.

The number of contamination found during the first period is in the Table 2.

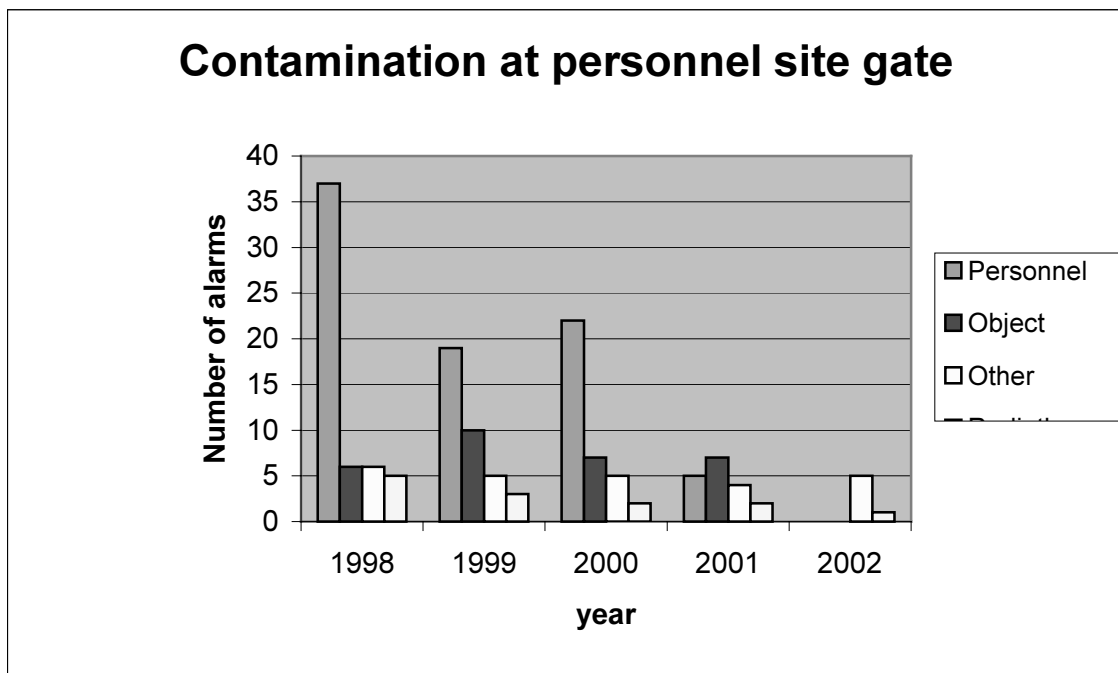


Table 2

Year	No of personnel contamination	No of object contamination	Other	Radiotherapy
1998	37	6	6	5
1999	19	10	5	3
2000	22	7	5	2
2001	5	7	4	2
2001-2002 after the modernisation	0	0	5	1

The contamination events were divided into the four groups:

1. Personnel contamination – the group contains all the events caused by worker’s surface body contamination, his/her dressing or internal contamination.
2. Object contamination – contamination of objects, things wearing by personnel.
3. Other – contamination found at the entry to the NPP site which are brought from other sites or caused by natural contamination.
4. Radiotherapy – usually found at the entry.

Installation of the monitors PM 7 at the personal site gates, which replaced the previous old insufficient monitors, had the large influence into management of the radiation contamination control at the exit from RCA as well as to the evaluation of the radiation knowledge of the plant personnel, contractors and visitors. The RP training process had been analysed and improved and personnel was provided by relevant information. This is why the number of contamination events during the following years decreased.

Analysis of the events revealed:

- More than 70% events of personnel contamination was caused by insufficient characteristics of old equipment RZB-04-04 a RUSSI-1 at the exit from RCA.
- Less than 30% of personnel contamination was caused by human error – by violation of RP rules at the exit from RCA.
- Up to 60% of events was caused by the contractors workers.
- Up to 80% of all events was found at JE V-1 which was caused by a large extend of reconstruction works during the reconstruction and the higher contamination of the primary circuit.
- All contamination events represent only negligible risk on the human health (the maximum found activity moved around the kBq).

After the 1st June 2001 no exceeding of activity alarm due to the real contamination has been registered at personal site gates of Bohunice NPP. The fact can be explained by higher radiation

awareness of the personnel and mainly by finishing of the replacing of the personal contamination monitors at the exit from radiation controlled areas.

Conclusion

The first event of contamination at the personal site gates was found after the installation of monitors PM 7. The main cause of the taking out of the contamination from the RCA was the insufficient monitoring equipment at the RCA boundary. The problem was eliminated by replacing old monitors through new ones – APM 3A and by the completing of CPO monitors.

RADIOLOGICAL ASSESSMENT OF RADIOACTIVE CONTAMINATION ON PRIVATE CLOTHING

F. Schartmann, S. Thierfeldt
Brenk Systemplanung GmbH, Germany

Abstract

In the very rare cases where private clothing of persons working in a nuclear installation are inadvertently contaminated and this contamination is not detected when leaving the facility, there may be radiological consequences for this person as well as for members of his or her family. The VGB (Technische Vereinigung der Grosskraftwerksbetreiber) in Germany has detailedly investigated the spread of contamination in nuclear power plants. Part of this evaluation programme was a radiological analysis which has been carried out by Brenk Systemplanung GmbH (Aachen/Germany).

The radiological analysis started with the definition of the source term. It is highly unlikely that activities of more than 5 kBq ^{60}Co could leave a plant undetected on the body or the clothes. Nevertheless activities up to 50 kBq and different nuclide vectors were regarded. It has been found that ^{60}Co is the most important contaminant.

The radiological analysis focusses on two types of contamination: particles and surface contamination. The pathways by which such a contamination can lead to an exposure by external irradiation or by ingestion depend on the type of contamination and are analysed in detail. For example, a particle could be retained in pockets or other parts of clothing and may lead to prolonged external irradiation until the piece of clothing is washed. The analysis is performed on the basis of conservative to realistic assumptions. It yields the following results (based on deterministic and probabilistic models):

- Surface contamination may lead to only small doses in the trivial dose range (a few tens of μSv per incident).
- The doses which may result from particle contamination may generally be higher. In rare cases the doses may exceed the trivial range, doses of a few mSv (per incident) cannot be excluded.

In conclusion, the analysis has shown that especially particle contamination needs to be focussed on. However, by the advanced detection equipment in German plants doses which may pose a health hazard can safely be excluded.

Introduction

In very rare cases the contamination of private clothing of persons leaving the controlled area of a nuclear installation may not be detected. Thereby radiological consequences for this person and the members of the family may occur. A radiological calculation was carried out by Brenk Systemplanung GmbH (Aachen/Germany), taking into account different aspects like source terms, nuclide vectors, pathways leading to contamination and exposure scenarios. The dose rates are calculated with deterministic and probabilistic models and the results are compared.

This paper relates to the paper “Radioactive Contamination on Private Clothing” of Manfred Meyer of Kernkraftwerk Philippsburg.

Source terms for contamination on private clothing

Contamination Mechanism

A thorough evaluation of possible pathways by which activity on clothing may eventually be transferred from the controlled area to a place outside has been performed by the German power utilities and has revealed the following main possible pathways:

- material lock: from material which is brought out of the controlled area part of the contamination may be transferred to persons, other material, vehicles;
- exit of the controlled area: a particle or surface contamination may remain on the body of the person leaving the controlled area. This contamination may be hidden on a part of the body where it is shielded and not detected by the person monitor;
- small objects: small items which a person takes out of the controlled area may bear some contamination which is not detected in the gate monitor.

Types of contamination and source term

In order to develop a radiological model, it is important to take into account the different nature of types of contamination which may be present on private clothing. Contamination may have the form of particles or of surface contamination. In the first case, the contamination is very localised and may be transferred only in the form of the entire particle, in the latter case it is distributed over a larger area of the clothing and may be transferred in part or in total. Both types of contamination have been taken into account in the radiological assessment.

Particle contamination

Particles are small pieces of matter which are highly insoluble and which possess a comparatively high specific activity. They may be transferred only intact and do not stick firmly to the clothing. Instead, they may fall off and be transferred to other places where they could lead to exposure. On the other hand, they may stick e.g. in pockets, pleats or seams of garments for longer times than surficial contamination which is totally dissolved when being washed.

Particles with a high ^{60}Co activity often originate from steels with high Co contents. Radioactive decay leads to electrical charging of the particles making them highly mobile. This is accounted for in the radiological scenarios.

For the assessment it is assumed that particles will contain maximum activities of 50 kBq ^{60}Co . No case of particle contamination in German NPPs with activities of more than 5 kBq ^{60}Co has been reported. However, in order to be enveloping, a maximum activity of one order of magnitude higher has been assumed.

Those particles described here are generally too large to be inhaled. On the basis of a density of 3.5 g/cm^3 and specific activities in the range between 10^7 and 10^9 Bq/g , diameters in the range of $70 \mu\text{m}$ for 5 kBq and $320 \mu\text{m}$ for 50 kBq are calculated whereas particles are inhalable only with diameters of less than about $20 \mu\text{m}$.

Surface contamination

Surface contamination is regarded here as a more or less homogeneous contamination on the clothing which is distributed over several 100 cm^2 . The contamination mechanism will usually involve contaminated liquids or greasy contaminated material surfaces with which the piece of clothing has come into contact. Part of the surface contamination can be transferred to the skin or be incorporated.

For the radiological assessment, an activity of 5 kBq ^{60}Co has been chosen. It has been assumed that the contamination is distributed over 400 cm^2 .

Nuclide vectors

The radiological assessment (section 0) has been carried out individually for a number of nuclides which might be present in NPPs as well as for typical nuclide vectors. For the sake of space, only the results for ^{60}Co and for 2 typical nuclide vectors are reported in this paper (table 1). NV1 consists only of the most important β/γ emitting nuclides ^{60}Co and ^{137}Cs while NV2 is more representative for the actual contamination composition including around 1% of α emitting nuclides as well as pure β emitters.

Table 1. Nuclide vectors for the radiological assessment

Nuclide	^{60}Co	^{63}Ni	^{65}Zn	^{90}Sr	^{137}Cs	^{154}Eu	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{241}Am	^{242}Cm	^{243}Cm	^{244}Cm
NV1	50%				50%									
NV2	40%	4%	10%	5%	30%	10%	0.1%	0.1%	0.1%	0.3%	0.1%	0.1%	0.1%	0.1%

The radiological assessment

Basic approach

The radiological assessment was carried out in the following steps:

1. Pathways leading to contamination of private clothing (section 0).
2. Source term and nuclide vectors (section 0 and 0).
3. Exposure scenarios (section 0).
4. Dose calculation (section 0).

The main part of the assessment was carried out using deterministic scenarios, i.e. scenarios with a fixed set of parameter values describing enveloping exposure situations. The results of such an assessment are doses which are highly unlikely to underestimate the exposure in the given circumstances. In order to get a more realistic assessment, a probabilistic assessment was also carried out. In this case, parameter values are left to vary according to predetermined statistical distributions and between upper and lower boundaries thus representing variation in real-life situations. Probabilistic analyses can help to assess the probability with which the conservative estimates of the deterministic assessment might occur.

Description of scenarios

Radiological scenarios establish the link between the activity and the resulting dose. A number of scenarios have been chosen to describe situations in which a person is exposed by particle or surface contamination, taking the various transfer mechanisms into account. Where appropriate, the age groups according to ICRP are considered: 0-1 a, 1-2 a, 2-7 a, 7-12 a, 12-17 a and > 17 a (adults). The dose evaluation is performed for all age groups and the maximum value is taken.

Scenarios for surface contamination

- A1: Contamination on a jacket over 400 cm². The jacket is worn for 30 days for 5 h each day.
- A2: Initial contamination on the hand of a person. 5 % of this contamination is inadvertently ingested. The rest remains on the hand for 24 h until being washed off (skin contamination).
- A3: Surface contamination is transferred to the private clothing of a person while being on the site of a nuclear power plant (outside controlled area). At home, this person is in contact with a small child which inadvertently ingests 50% of this contamination.
- A4: Initial contamination on the hand of a person. At home, 50% of this contamination is transferred to an item (e.g. curtain) in the sleeping room where it remains for 1 a and leads to external irradiation.

Scenarios for particle contamination

- B1: Particle on/in a jacket. The jacket is worn for 60 days for 5 h each day.

- B2: Particle contamination is transferred to the private clothing or the shoes of a person. At home, this particle is transferred to the hands and is then inadvertently ingested.
- B3: Particle contamination is transferred to the private clothing of a person while being on the site of a nuclear power plant (outside controlled area). At home, this particle is transferred to a piece of furniture where it remains for 1 month and leads to external irradiation.
- B4: Particle contamination on a small private object, e.g. spectacles or watch. This object is assumed to be worn 16 h per day. The particle remains for 1 month and then falls off.

Results

Dose calculations

The results of the dose calculations are presented for ^{60}Co (Table 2) and the two nuclide vectors presented in section 0 (Table 3 and 4). All calculations refer to 5 kBq for surface contamination and 50 kBq for particle contamination. This activity relates to γ emitting nuclides.

Table 2. Dose calculations for ^{60}Co

<u>Surface Contamination</u>					
Parameter	Scenario:	A1	A2	A3	A4
(critical) age group		>17a	12-17a	1-2a	<1 a
dose from external irradiation	μSv	63,75	10,20	0,00	0,42
dose from skin contam., effect. dose	μSv	0,02	0,06	0,00	0,00
dose from ingestion	μSv	0,00	3,95	67,50	0,00
sum of doses	μSv	63,8	14,2	67,5	0,4
<u>Particle Contamination</u>					
Parameter	Scenario:	B1	B2	B3	B4
(critical) age group		12-17a	1-2a	<1 a	>17a
dose from external irradiation	μSv	2136,00	0,00	3,32	797,44
dose from ingestion	μSv	395,00	1350,00	0,00	0,00
sum of doses	μSv	2531	1350	3,3	172

Table 3. Dose calculations for NV1

<u>Surface Contamination</u>					
Parameter	Scenario:	A1	A2	A3	A4
(critical) age group		>17a	12-17a	1-2a	<1 a
dose from external irradiation	μSv	39,38	6,30	0,00	0,26
dose from skin contam., effect. dose	μSv	0,02	0,07	0,00	0,00
dose from ingestion	μSv	0,00	5,23	48,75	0,00
sum of doses	μSv	39,4	11,6	48,8	0,3
<u>Particle Contamination</u>					
Parameter	Scenario:	B1	B2	B3	B4
(critical) age group		12-17a	1-2a	<1 a	>17a
dose from external irradiation	μSv	1344,00	0,00	2,09	501,76
dose from ingestion	μSv	522,50	975,00	0,00	0,00
sum of doses	μSv	1867	975	2,1	108

Table 4. Dose calculations for NV2

<u>Surface Contamination</u>					
Parameter	Scenario:	A1	A2	A3	A4
(critical) age group		>17a	12-17a	1-2a	<1 a
dose from external irradiation	µSv	34,82	5,57	0,00	0,23
dose from skin contam., effect. dose	µSv	0,02	0,06	0,00	0,00
dose from ingestion	µSv	0,00	6,56	60,52	0,00
sum of doses	µSv	34,8	12,2	60,5	0,2
<u>Particle Contamination</u>					
Parameter	Scenario:	B1	B2	B3	B4
(critical) age group		12-17a	1-2a	<1 a	>17a
dose from external irradiation	µSv	1186,76	0,00	1,85	443,06
dose from ingestion	µSv	656,08	1210,34	0,00	0,00
sum of doses	µSv	1843	1210	1,8	95,6

As can be seen from Table 3 and 4, the results are largely dependent on the strong γ emitters because external irradiation will play the most important role in all scenarios (for nuclide vectors which are typical for NPPs). The results do not differ significantly between NV1 and NV2. Furthermore, a comparison with Table 2 reveals that it is obviously an enveloping approach to attribute the entire contamination to ^{60}Co which indeed has been found to be the most important nuclide in real contamination events.

Comparison with dose criteria

The results obtained in section 0 have to be judged against dose criteria. Possible dose criteria are:

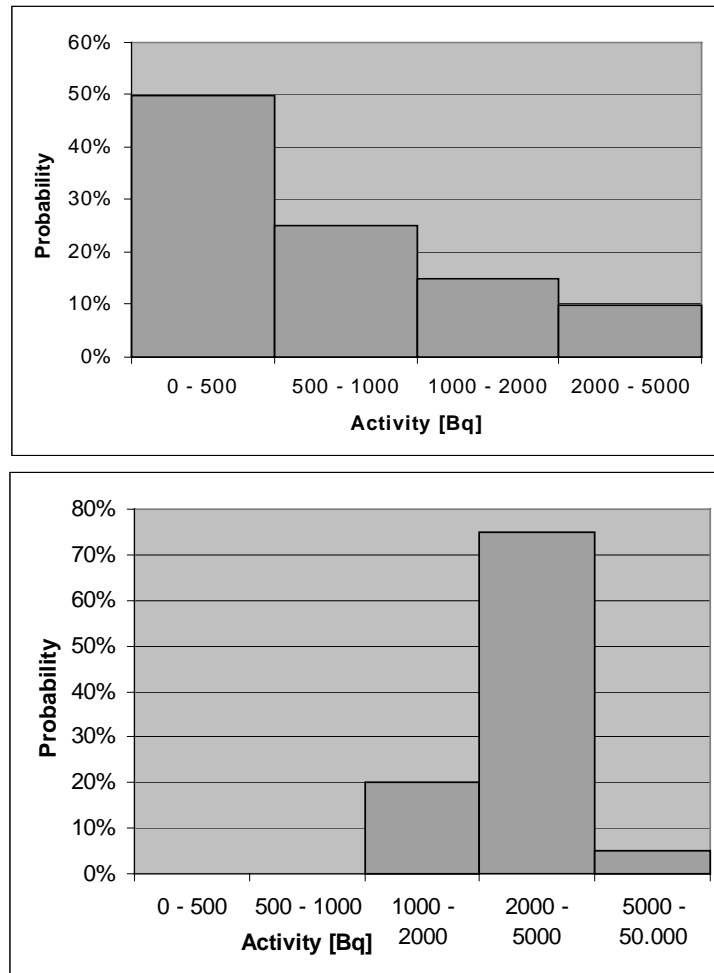
- The range 10 to 100 µSv which is usually regarded as the trivial dose range. Doses not exceeding this range could be regarded as negligible.
- 1 mSv: 1 mSv/a is the annual dose limit for members of the general public from practices according to the Basic Safety Standards of the European Union and of the IAEA. Doses in this range (from one single incident of contamination of private clothing) could therefore be regarded as acceptable if it is highly unlikely that several incidents per year might occur for the same person.
- 6 mSv/a / 20 mSv/a: these dose levels apply to controlled workers and are therefore not applicable to exposure situations of members of the general public.

The result of the radiological assessment shows that any dose which could reasonably be expected from a single incident lies below around 100 µSv for surface contamination and up to about 2 mSv for particle contamination. That means that surface contamination will in any case lead to only negligible doses while particle contamination has the potential also for doses beyond the trivial range. This result, must be judged in light of the fact that up to now only particles with a tenth of the activity which has been assumed here have been detected. Therefore, it is highly unlikely that also in the case of particle contamination on private clothing in Germany doses have been above the 100 µSv range.

Additional probabilistic assessment

As mentioned in section 0, a probabilistic assessment has also been carried out in which the most important parameter values were allowed to vary according to predetermined statistical distributions. From actual cases, the probability distributions for the activity in the case of surface contamination and of particle contamination shown in Figure 1 have been derived.

Figure 1. **Probability distribution for the activity for surface and particle contamination**



Other parameters allowed to vary were:

- the probability that a child would be affected at all in scenarios A3 and B2 according to German demographical data;
- the time a garment is worn;
- transfer factor to the skin or to curtain/furniture;
- transfer factor of surface activity being inadvertently ingested;
- distance to the source (for calculation of external irradiation);
- and others.

Typical results for scenarios A3 and B1 for ^{60}Co are shown in figure 2. It can be seen e.g. for scenario B1 that it can be expected that the most probable doses will not be in the range of 2 mSv as calculated in the deterministic approach but may rather lie in the range of a few 100 μSv . The statistical data for the results are given in Table 5.

Figure 2. Results of probabilistic dose calculations for ^{60}Co for scenarios A3 and B1 (note: different ranges on abscissa)

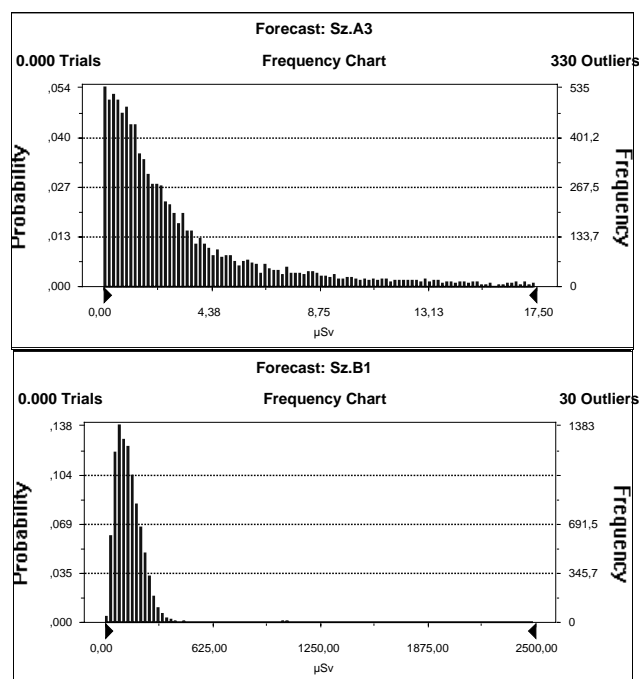


Table 5. Statistical data for the probability distribution of the doses for scenarios A3 and B1 for ^{60}Co

Percentile	Scenario B3 [μSv]	Scenario B1 [μSv]
50% (Median)	2,01	133,86
95%	14,28	359,66
97,5%	19,48	1110,84
Maximum	53,62	3903,40
Comparison deterministic	67,5	2531

Conclusions

The comparison of the deterministic and probabilistic calculation methods reveals the following aspects:

- The deterministic calculations or results are obviously conservative compared to the probabilistic calculations.

- The effective doses, which will occur in reality because of a contamination of private clothes (probabilistic calculations) are around one order of magnitude smaller than the deterministically determined doses.
- It is very improbable that in case of surface contamination the effective dose will reach the trivial dose range of a few 10 μSv up to 100 μSv . But in cases of particle contamination effective doses of a few 100 μSv (95%-percentile) or up to the mSv-range (97,5%-percentile) can be reached.

That means that surface contamination will in any case lead to only negligible doses. In case of particle contamination it is highly unlikely that contamination on private clothing in Germany has effected doses exceeding the 100 μSv range.

In conclusion, the analysis has shown that especially particle contamination needs to be focussed on. However, by the advanced detection equipment in German plants doses which may pose a health hazard can safely be excluded.

RADIOACTIVE CONTAMINATION ON PRIVATE CLOTHING

M. Meyer

EnBW Kraftwerke AG, Germany

Introduction

In 1999 during the annual outage of a German nuclear power plant (NPP) radioactive contamination up to 5 KBq (^{60}Co) was found on private clothing of arriving contractor personnel. The contamination on private clothing was found by check-in measurements at the reception, before entering the controlled area. This contamination could only be found as a new type of sensitive Fast-Scan-Monitor (whole body incorporation and contamination monitor) with high efficiency was used.

The Federal Ministry for Environment, Nature Conservation and Nuclear Safety (BMU) was informed about this incident through correspondence by the State Ministry of Environment of Lower Saxony, that itself was informed by the plant operator. The BMU demanded from the state authorities for further information, clarification and a statement.

To evaluate how this kind of contamination spread is possible to gain out of radiation controlled areas (RCA) of nuclear power plants, VGB (Technical Association of Large Power plants Operators/Essen) established a working group called "Contaminated Clothing".

By using a review program the working group investigated on a selected number of German nuclear power plants, through a review, the possibilities of contamination spread beyond the boundaries of radiation control areas (RCAs). The review team was given a mandate to identify existing program deficiencies and recommend corrective measures.

The results of the reviews in substance were completed by the statements from different independent Technical Surveillance Agencies (TÜV), and exist as recommendations.

For all German NPPs the review experience and additional statements of TÜV resulted in improvement measures, which were introduced and discussed with the state authority. The implementation of measures is specific to each plant, because there is no uniform administrative, organisational and technical standard in German NPPs on this item.

To assure the review results and to look for radiological consequences for personnel or civilian population, an independent study was given by contract to "Brenk Systemplanung" company. The intention was to make a radiological evaluation of contamination on private clothing with deterministic and probabilistic methods, and to show that there is no significant radiological consequence.

Paths of contamination spread

Although there is no standardised procedure in German NPPs to avoid spread of contamination by personnel, the administration, organisation and technical preconditions are comparable in all German nuclear power plants. The essential barriers to avoid contamination spread in NPP's by leaving the radiation controlled area are noted below as:

- whole body contamination monitors for personnel;
- gamma monitors (release measurement device/chamber) for all personal materials/small objects;
- radiation and contamination control in lorry and materials locks only by personnel from radiological;
- protection department;
- release of residual materials only by beta and gamma monitoring (release measurement device).

The essential barriers for contamination spread in nuclear power plants by leaving the surveillance (not installed in all German NPPs) area are the following:

- whole body gamma check at the entrance and exit for personnel;
- contamination gamma scan monitors for vehicles;

Only through these paths spread of contamination is possible.

In spite of good technical and administration preconditions to survey contamination spread, contamination was found on private clothing of contractor personnel by check-in measurements, before entering the controlled area.

Contamination form of appearance

Basically one has to consider two contamination scenarios for spread of contamination.

Surface contamination, which is widely a homogeneous contamination and the hot particle contamination. Hot particles are small, loose, highly radioactive particles. These particles are highly transportable due to their small size and electrostatic charge.

Up to now detected and documented incidents of surface contamination outside of radiation controlled areas indicate a good consistence to all available data sources, which shows, that contamination on private clothing or skin is generally in a total height in the range of 100 to 1 000 Bq. The surface which was recognised by this incidents was always less 1 000 cm², a typical value is about 200 cm².

For hot particle contamination outside of radiation controlled areas only a few incidents are noted. Today's standard of knowledge shows one single incident where a hot particle contamination was found on a cardigan (jacket) with 5 000 Bq ⁶⁰Co. Some other cases, but only few, with hot particle contamination are noted in front of vehicles and material locks with the same height of contamination.

Review team methods and goals

The explosiveness of this subject and the information of BMU disposed VGB to establish a team of German health physics experts, with the mandate, to investigate by Peer Review – on selected German nuclear power plants – the technical and administration realities (preconditions) of radiation protection.

On the basis of the Peer Reviews a catalogue of recommendations was issued to reduce the potentiality of contamination spread. For recommendation this catalogue was given to all German nuclear power plants to check their own procedures, standards and programs.

During the Peer Review the team acts on always the same criterions, as noted below.

Achievement of the following goals

- improvement of standards for the supervision of contamination, open exchange of experience, mutual;
- comparison and examination of individual selected procedures;
- review of contamination prevention concepts in RCAs;
- analysis of selected, particular contamination control processes;
- identification of good practices and weaknesses;
- identification of organisational and administrative improvement possibilities;
- elaboration of proposals and recommendations;
- identification of contamination paths (sources, material flow, personnel flow).

Composition of the Review Team

- 1 teamleader (no staff member of the plant);
- 1 keeper of the minutes;
- 2 health physics experts.

Review preparation

Elaboration of a list of places and items (working places, hot workshop, storage depots, decontamination areas, vehicle locks, material locks, personnel and material flow) should be checked by the team experts. Based on this list a quick familiarisation to preconditions and comparability of findings in different plants should be ensured. During preparation the catalogue could be adjusted to plant specific conditions.

Review performance

For evaluation of procedures the review team applies 5 methods:

- inspection of documents;
- discussion with operational staff on site;

- discussion amongst the procedure experts;
- observation of procedures in practise;
- discussion of findings with operational staff on site.

Review duration is 3 days

Considerations and exchange of experience

Good practices and weaknesses will be discussed. In case the majority of team members agrees to individual findings, these have to be considered in more detail. Therefore the teamleader nominates a keeper of the minutes. He is responsible to draft a short report describing the weak points identified, possible reasons and practical proposals for improvement.

The head of radiation protection has to be informed daily and the plant manager has to be informed by final report.

Final report

During the last day the final report has to be elaborated under the guidance of the teamleader. The report contains a short description of all items considered description of good practices and the presentation of weak points including reasons (if possible).

Referrals of the review team

Administrative measures

The responsibility for the keys that lock the rooms adjacent to the RCA (i.e. RCA entree, and the various locks) should be excessively limited to radiation protection personnel.

The lobby area adjacent to the RCA must be kept free of contamination. This includes that contaminated material which was used in the RCA, must be denoted with the appropriate signs available.

A routine measurement of the contamination level is necessary. It is excessively important to use cleaning devices free of any contamination to clean the area adjacent outside to the RCA (i.e. RCA entrees and locks) These cleaning devices should be likewise checked on a routine basis for any source of contamination.

This also includes rigging loops, slips, ropes and traverses that are in use in the locks adjacent to RCA. These machines, materials etc, should be denoted after the routine check.

Private clothing should absolutely not be permitted inside the RCA.

The area around the RCA exit should be kept under constant surveillance by radiation protection staff, by personnel flow or materials brought out of the RCA.

It is necessary to ensure contamination prevention measures on the wheels of transportation cars that are used in and outside of contamination zones. (i.e. decontamination, covering the wheels with tape or adhesion foil).

Monitoring measures

The background level on contamination monitors for measuring personnel and objects should be kept under an adequate level. This is achieved with a dose rate below 100 nSv/h.

The contamination monitors should be calibrated with the “guide/dominant nuclide”, to determine the efficiency and detection limit.

The necessary contamination control on tools, paper etc. must additionally be monitored with an integral gamma measurement.

Additional control measures

Occasional contamination monitoring should be carried out to recognise any possible contamination spread in the areas adjacent to the RCA (i.e. airlocks, dressing rooms) by random tests.

The materials that have left the RCA should be occasionally monitored by random tests.

The radiation protection clothing that is used outside the RCA (i.e. spent fuel cask processing) should be checked on radioactivity. This also applies to vehicles leaving the plant vicinity.

The tools and rigging loops used in the workshop inside the RCA, are to be monitored routinely with contamination control checks.

Procedures in event of contamination findings

If any contamination is discovered on private clothing, the following measures should be taken:

- detection of β/γ contamination by monitoring;
- initiate an inquiry for clarification reasons;
- induce further measures according to the findings.

Summary

The organisational and administrative measures for contamination prevention and the technical methods for contamination control, stated in the review for the different plants are comparable, and results, in accordance to the German standards, indicate a high state of administrative, organisational and technical precaution.

A fundamental modifications of the procedures are not necessary. The measures performed are for optimational reasons.

It should be pointed out, that undiscovered contamination (i.e. unrepresentative skin surfaces, contamination spread over a small area) in relation to radiation monitoring, over geometrical unfavourable spots, can never eliminate the problem to 100%. The probability can be further minimised through the supplementary installed measures.

PRACTICAL LIMITATIONS OF MONITORING PERSONAL CONTAMINATION AT A PWR

M.P. Lunn, G.A. Renn

British Energy Generation Ltd, United Kingdom

Abstract

Although personal contamination events are usually of low radiological significance, at Sizewell B they have often been subject to intense managerial & regulatory interest. A recent internal review of Sizewell B's arrangements for the assessment and recording of personal contamination events included a re-evaluation of the protocol for the calibration of installed personnel monitoring equipment. This paper summarises the findings of the study. The practical implications of the findings are discussed.

Introduction

Personnel contamination monitoring at Sizewell B

All personnel that enter radiological controlled areas (RCA) at Sizewell B are monitored for contamination before leaving. A variety of instruments are used, typically whole body monitors, although hand & foot monitors or hand held probes are occasionally employed for low occupancy, low risk RCAs external to the main reactor block.

At the main exit from the RCA, and also in change facilities adjacent to areas where a significant risk of personal contamination exists, NE Technology IPM8 whole body monitors are used to detect contamination on clothing and skin.

Sizewell B, in common with the majority of UK nuclear establishments, normally uses a derived working level (DWL) for beta-gamma surface contamination of 4 Bq/cm², for the clearance of personnel and equipment from the RCA. This value derives from long-standing custom and practice [1].

Calibration of installed personal monitors

Original protocol

Historically activation products of steel, primarily ⁶⁰Co, ⁵⁸Co, ⁵¹Cr and ⁵⁴Mn, have been the dominant radioactive contaminants at Sizewell B. Operational contamination instrument derived

working levels have been determined using the assumption that any contamination detected is due to ^{60}Co .

The long-standing calibration procedures for installed contamination monitoring instruments at Sizewell B have used planar ^{36}Cl sources, with an assumed P-factor of 2. Alarm set points are calculated using equation 1 below [2].

$$\text{(Equation 1)} \quad C = \frac{Aa\Sigma}{P}$$

Where: C is the alarm level (cps).
A is the DWL (Bq/cm^2).
a is the area over which the contamination is averaged (cm^2)
 Σ is the detector efficiency as a proportion of unity.
P is the P-factor relating surface activity to surface emission.

The detector efficiencies are determined with the sources in contact with the detector. Additional *ad-hoc* checks have been performed on random detector panels using ^{55}Fe and ^{99}Tc sources, to confirm expected response to electron capture nuclides and low energy beta emitters.

To take account of the fact that the beta maximum energy of ^{36}Cl (E_{max} 714 keV) is greater than that of ^{60}Co (E_{max} 310 keV), Sizewell B originally set the installed personnel monitors to alarm at 2 Bq/cm^2 , thus compensating for the lower detector efficiencies at ^{60}Co beta energy.

The review of calibration procedures recognised that it would be preferable to use calibration sources closer in energy to the isotopes of interest. Use of ^{60}Co as a calibration standard was ruled out because its relatively short half-life (5.27y) would make it necessary to replace calibration sources on a regular basis. Technetium-99 was considered because it has a maximum beta energy (E_{max} 290 keV) close to that of ^{60}Co and a long half-life (2.12×10^5 y). Certainly this nuclide is typically used for instrument calibrations on US light water reactors similar to Sizewell B.

However it was judged that, taking into account self- absorption and attenuation in clothing, the energy spectrum of ^{60}Co would in fact be closer to that of ^{14}C (E_{max} 156 keV). For this reason and also because of its long half-life (5 760y) and the ready availability of traceable standards for periodic source re-calibration, ^{14}C was eventually selected as the new calibration standard.

Study protocol

Detector efficiencies

The variability in detector – clothing distance for various body parts was determined using 14 members of staff. Based on these measurements, ^{14}C efficiencies were determined using 100 cm^2 activated foils, traceable to the UK National Standard. Background & foreground counts were determined over 100 seconds.

Setting an alarm level

Based on these efficiencies, various alarm levels were calculated & assessed for their practical application.

IPM8 response to contaminated clothing

The response of an IPM8 when presented with contaminated clothing was compared for alarm settings derived from calibrations based on ^{14}C and ^{36}Cl . 100 cm² cotton patches (fabric weight 12 mg/cm²) were spiked with known amounts of ^{60}Co , ^{63}Ni or ^{54}Mn solution. The activity on each patch was confirmed by hand-held probe & γ - spectrometry (where appropriate).

These were then backed with polythene (to prevent spread of contamination) and worn on various body parts (Table 1) through an IPM. Each patch was passed through the IPM 3 times, and the number of alarms was recorded. The minimum detectable activity (MDA) was defined as lowest activity patch that could generate 2 out of 3 alarms.

Table 1. Locations where contaminated patches were worn

Front	Rear
Lower shin	Lower calf
Knee	Knee
Thigh	Thigh
Waist	Buttocks
Abdomen	Centre of lower back
Chest	Middle of shoulder blades

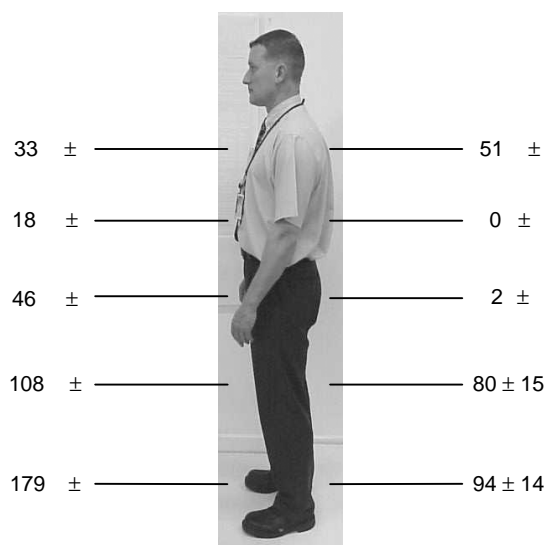
Ten second counts were used throughout the patch tests, as this is the count time used operationally. Data is shown as mean \pm 95% confidence levels where appropriate.

Results

Clothing – Detector separation

A great deal of variability in the detector-clothing distance was observed both between individuals and also along an individual's body. The data is shown in Figure 1.

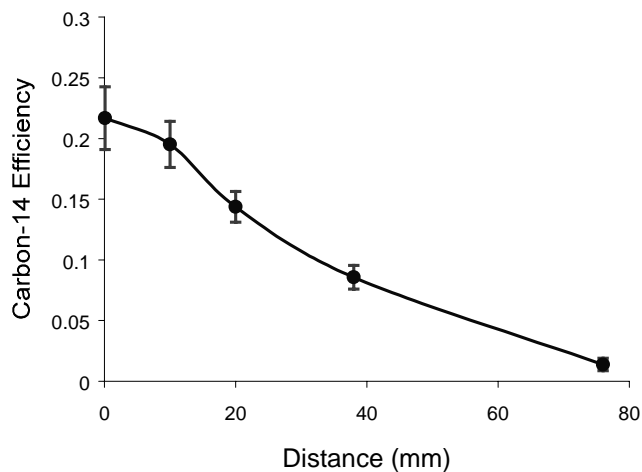
Figure 1. **Clothing – Detector distances (in mm) for various anatomical regions**



Carbon-14 efficiencies

Figure 2 shows the efficiency of an IPM8 body panel to ^{14}C at various source – detector distances. Contact efficiencies (e.g. representing the front abdomen, buttocks etc.) are $22 \pm 3\%$. At a distance of 80 mm (e.g. the knees) the efficiency has fallen to $1.4 \pm 0.5\%$.

Figure 2. **Variation in detector efficiency to ^{14}C with increasing source-detector distance**



From this data it is obvious that calibrating personal contamination instruments based on contact efficiencies was unrealistic because, at Sizewell B, the body parts that are most often contaminated (apart from the hands and feet) are the knees and ankles.

Various alarm levels were calculated for the efficiencies determined in Figure 2. The lowest alarm level that could be set for reliable operation was based on the efficiency derived at 20 mm. Alarm levels based on efficiencies at distances greater than 20mm led to continuous high background or contamination faults.

Response to contaminated clothing

The characteristics of each contaminated cotton patch are shown in Tables 2 and 3.

Table 2. Spiked 100cm² cotton patches – ⁶⁰Co

Nominal activity (Bq/cm²)	Net cps by Eberline HP260 hand-held contam'n probe	Activity determined by γ – spectrometry (Bq)
2	2-4	167 ± 22
4	4-6	317 ± 31
8	6-10	687 ± 49
20	12-18	1651 ± 83
40	30-40	3 220 ± 148

Table 3. Spiked 100cm² cotton patches – ⁵⁴Mn

Nominal Activity (Bq/cm²)	Net cps by Eberline HP260 contamination probe	Activity determined by γ - Spectrometry (Bq)
2	< 0.5	183 ± 22
4	0.5-1	327 ± 31
8	1-2	681 ± 49
20	2-3	1 535 ± 79
40	3-4	3 203 ± 138

The hand-held probe response to ⁶⁰Co was approximately 10 times greater than the response to ⁵⁴Mn contamination at the same specific activity (see Tables 2 and 3). No response was observed for any patch spiked with ⁶³Ni.

The minimum specific activities required to generate at least 2 out of 3 alarms are shown in Figures 3 and 4. For both radionuclides there is a significant variation in the MDA between the front and rear of the body and along the vertical axis. Generally, the lowest MDAs were found on the rear torso whilst the highest MDAs were observed on the front lower legs.

Figure 3 shows that in the case of ^{60}Co , alarm settings based on the ^{36}Cl calibration protocol were typically in the range of 20 – 40 Bq/cm^2 (i.e. 5 to 10 DWL), although an MDA of 8 Bq/cm^2 (2 DWL) was achieved in the small of the back. Calibration with ^{14}C & the resultant [reduced] alarm levels improved greatly the MDA for ^{60}Co . For example the MDA for the front waist was reduced from 20 Bq/cm^2 to 4 Bq/cm^2 . However, despite the reduction in alarm levels, the MDA for the legs still remains at 5 to 10 DWL.

Similar results were observed for ^{54}Mn (Figure 4). The MDA for each body region was typically twice that of ^{60}Co . There was no response of the IPM8 to any patch spiked with ^{63}Ni , regardless of body region tested.

Figure 3. **Minimum specific activity (Bq/cm^2) of ^{60}Co to generate at least 2 out of 3 alarms on an IPM8. Data in bold refers to ^{14}C based calibration. Data in parentheses refers to ^{36}Cl based calibration**

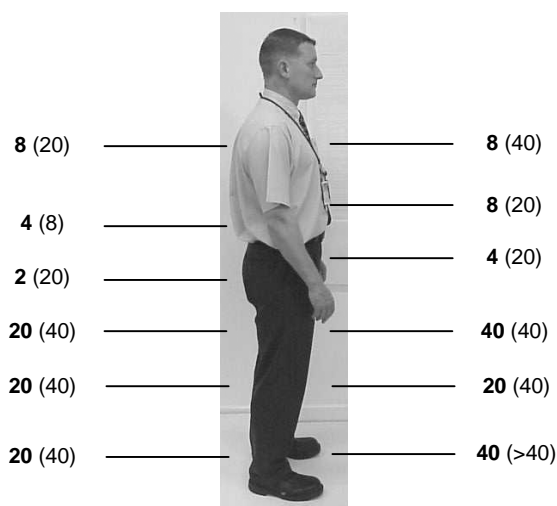
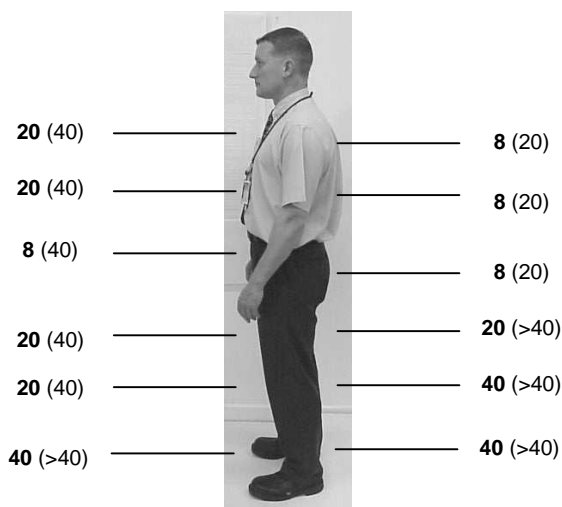


Figure 4. **Minimum specific activity (Bq/cm^2) of ^{54}Mn to generate at least 2 out of 3 alarms on an IPM8. Data in bold refers to ^{14}C based calibration. Data in parentheses refers to ^{36}Cl based calibration**



Discussion

National Physical Laboratory Good Practice Guide N^o 29 [3] has recently recommended that, when determining alarm levels on installed contamination monitors, the employer should use an efficiency at least a factor of 2 lower than the contact efficiency, in order to account for variable distances between clothing and the detectors.

In practice, when using ¹⁴C, we found that this recommendation equates roughly to the lowest practicable alarm level, despite the instruments being located in very low ambient background areas. Attempting to use lower alarm settings simply resulted in the instruments being in continuous “high background” fault alarm. Thus, the outcome of this calibration protocol revision has been a 5 to 8% reduction in the hand and foot alarm levels and a 50% reduction for the body panel detectors (summarised in Table 4).

Table 4. Summary of alarm settings

Detector	Alarm settings (cps)		% Change
	Old (³⁶ Cl based)	New (¹⁴ C based)	
Hand (β)	166	158	-4.8
Foot (β)	103	95	-7.8
Body (β)	41	20	-51.2

These reductions mean that it is now possible to achieve a ⁶⁰Co MDA of 2 to 4 Bq/cm² (i.e. 0.5 to 1 DWL) on anatomical regions that are in contact with the detector panel (e.g. buttocks, lower back, and abdomen). Although the ⁶⁰Co MDA for the lower legs is 40 Bq/cm² (10 DWL).

One implication of the study is that personnel may have been exiting the Controlled Area with contamination levels in excess of those allowed by Local Rules. Table 5 shows the personal contamination rate in the 12 months prior to, & immediately after, the alarm set point change; fortunately there is no evidence to suggest that this has been the case.

Table 5. Effect of alarm level change on PC event frequency

	Before (2000/2001)	After (2001/2002)
No. RCA Entries	85313	55334
No. PC Events	139	91
PC Event Rate (per 1000 RCA entries)	1.63	1.64

There are a number of reasons why the former set points were operationally adequate. Firstly in areas of significant contamination risk the operator usually passes through a number of diverse contamination monitoring points. Secondly, some personal contamination incidents involve multiple areas of the body. In these cases, it is contamination on the hands and feet (with very low MDAs) that is detected. Contamination on clothing with higher MDAs is then detected on the secondary survey by hand-held probe.

It is important to note that, from a practical radiological protection perspective, the release of personnel with levels of beta-gamma contamination between 4 and 40 Bq/cm² poses negligible radiological risk.

The reduction in the alarm level has not resulted in any increase in spurious alarms. Latterly Sizewell B has been experiencing some operational problems with failed fuel, with fission product activity being released to the Reactor Coolant System. Typical fission product maximum beta energies are higher than that of ⁶⁰Co, thus where fission product contamination is present then the revised alarm set points are conservative.

Despite these shortcomings, the installed personnel monitor is the instrument of choice for clearance monitoring of personnel. Installed monitors are responsive to a wider range of nuclides than many hand-held probes and they have far shorter monitoring times. Further improvements in the installed contamination monitor MDA could be achieved by design modifications. Improvements to the layout of detector arrays may yield lower MDAs but a more practical and cost-effective solution would be the introduction of individual alarm set points for each detector panel.

Conclusions

In order to detect personal contamination at levels that are required by Local Rules, we have revised our calibration procedures. Calibration of installed contamination monitors is now performed using ¹⁴C as the standard and as a result, the instrument alarm levels have been lowered by up to 50%. Even at this level, some parts of the body still have ⁶⁰Co MDAs in excess of the clearance criteria, thus highlighting the need for strict contamination control at the work place, before persons approach the RCA barrier.

Acknowledgements

The authors would like to thank Mike Barton for preparing the spiked cotton patches and performing the radiochemical analyses.

References

- [1] Dunster, H.J. (1955): Contamination of Surfaces by Radioactive Materials: The Derivation of Maximum Permissible Levels: *Atomics*, August, 233-239.
- [2] IPM8A Instruction Manual, NE Technology Ltd. (1993).
- [3] Pottinger, M. Woods, M. & Keightley, L. (2001); Measurement Good Practice No. 29, National Physical Laboratory, UK.

POSTERS

DOSE TRENDS ANALYSIS OF THE KRŠKO NPP

M. Černilogar Radež, H. Janžekovič, M. Križman
Slovenian Nuclear Safety Administration, Slovenia

Abstract

Since the beginning of the ISOE programme, the annual average collective dose per reactor covered by ISOE worldwide demonstrates downward trend and reached 0.96 man·Sv in 2000 [1] for PWR. The analyses of occupational dose trends in the period from 1990 to 2001 for the single Slovenian NPP at Krško are given with the emphasis in relation between the job and the doses. Although the use of the collective dose as a suitable parameter has a limited applicability it can be successfully applied in the analysis of these trends.

Introduction

The implementation of individual occupational dose limits of safety standards [2] related to occupational exposures requires a comprehensive system. In the frame of this system operators of nuclear installations, outside undertakings and state authorities should act in a harmonised way. According to Article 38 of the 96/29/EURATOM directive [3] a country should also make all necessary arrangements to recognise the capacity of the approved dosimetric services and qualified experts in order to assure the implementation of the standards taken from [3]. In addition, individual monitoring results are subject of Article 29 of the 96/29/EURATOM as well as the arrangements under which the result of individual monitoring are conveyed. In Slovenia approximately 25% of occupationally exposed workers are workers in nuclear installations [4]. In order to improve radiation protection in nuclear installations of Slovenia and following the requirements of the directive, the *Central Dose Register of Workers in Nuclear Installations* has been established at the Slovenian Nuclear Safety Administration [5].

The main parameter which can be used in order to estimate the influence of the specific practice on occupational exposure is usually the individual annual effective dose averaged over a set of all workers. In addition, also the values of maximum annual dose or median annual dose can describe the distribution of number of workers as a function of occupational doses. In some cases subsidiary dosimetric quantities should be used (collective equivalent dose, collective effective dose, dose commitment...). The concept of the collective dose defined in Paragraph 34 from [2] has been well used in ISOE reports and it is also used in [6] in contrast with the directive [3]. The collective dose can be a suitable parameter in order to compare the implementation of the safety culture at different nuclear installations. However, we should take into account the fact that the details of this implementation are hidden.

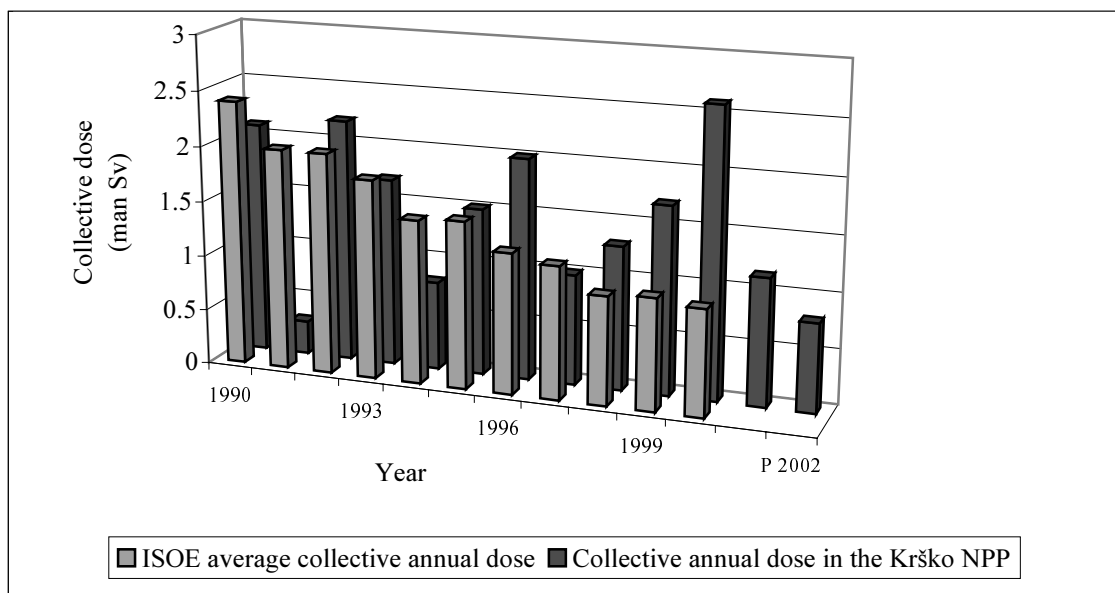
Analysis of dose trends in the Krško NPP

In the period from 1996 to 1999 the nuclear power plant Krško (Westinghouse pressurised water reactor with electrical output 700 MW) performed a comprehensive maintenance of the ageing steam generators (sleeving, eddy current testing, reactor pump replacement...). In 2000 both steam generators were replaced.

Figure 1 shows the trend of the annual collective dose in the period from 1990 to 2001 as well as the reported dose from ISOE report. Due to maintenance works, plant modification and upgrading the annual collective doses in the Krško NPP in the period from 1997 to 2000 show a rising trend just opposite to the trend reported in ISOE reports. In the year 2000 the collective dose reached the maximum value of 2.60 man Sv due to steam generators replacement and dropped down to 1.13 man·Sv in the year 2001. The major part of the dose reduction was related to the modernisation of the NPP and the reduction of outage duration. The minor part of the dose reduction could be also reached by the implementation of the work management principles (exchange of techniques and experiences in occupational exposure reduction).

The individual average effective dose in the year 2000 was 2.30 mSv and 1.27 mSv in the year 2001. These values represent less than 5% of the annual dose limit for the occupational exposure stated in legislation according which this dose limit is still 50 mSv. The internal annual dose limit of the NPP is 20 mSv. The expected annual collective dose of 0.8 man Sv in the Krško NPP in the year 2002 (*P 2002*) is also given in Figure 1.

Figure 1. The annual collective doses in the Krško NPP and the annual average collective doses per PWR taken from ISOE reports are given. The expected annual collective dose in the Krško NPP in the year 2002 is denoted by *P 2002*.



The analyses of doses from 1996 to 2000 point out that the major part of the collective dose is due to plant outage works (from 75 to 95%) while the contribution of the normal operation is much smaller. In the year 2000 approximately 70% of all workers were outside workers as they are defined in [7]. Those workers received approximately 87% of the total collective dose. The annual collective

doses in Krško NPP for plant personnel as well as for outside workers which were received during planned outages from 1996 to 2000 are given in Figure 2. The total collective doses during outages are also shown.

The collective dose of the plant personnel in the whole period showed quite constant values in contrast to the dose changes belonging to outside workers. This implicates that the special attention should be given to outside workers.

Figure 2. The annual collective doses in the Krško NPP for plant personnel as well as for outside workers in Krško NPP which were received during outages from 1996 to 2001. The total collective doses during planned outages are also given.

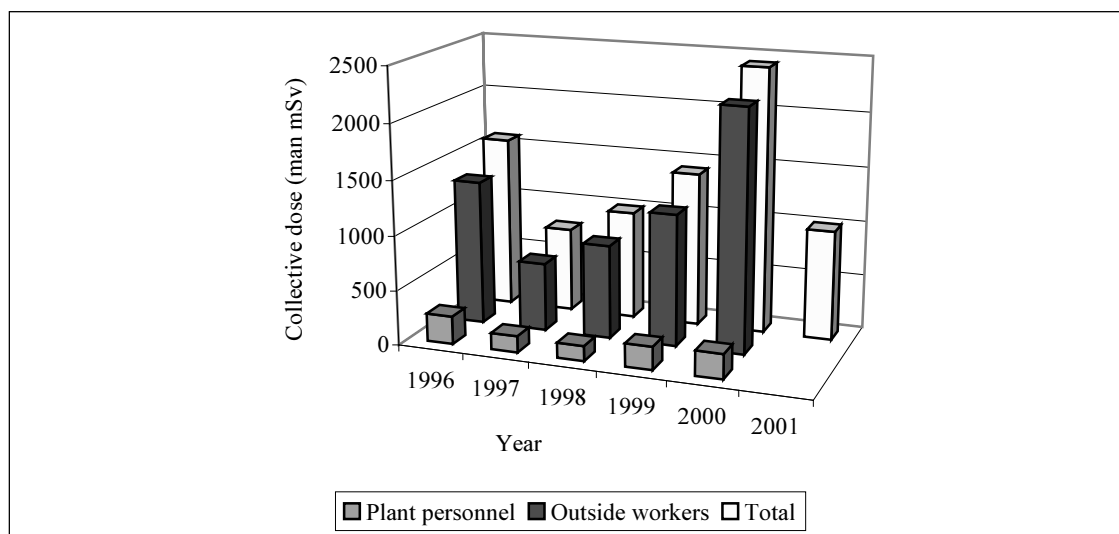
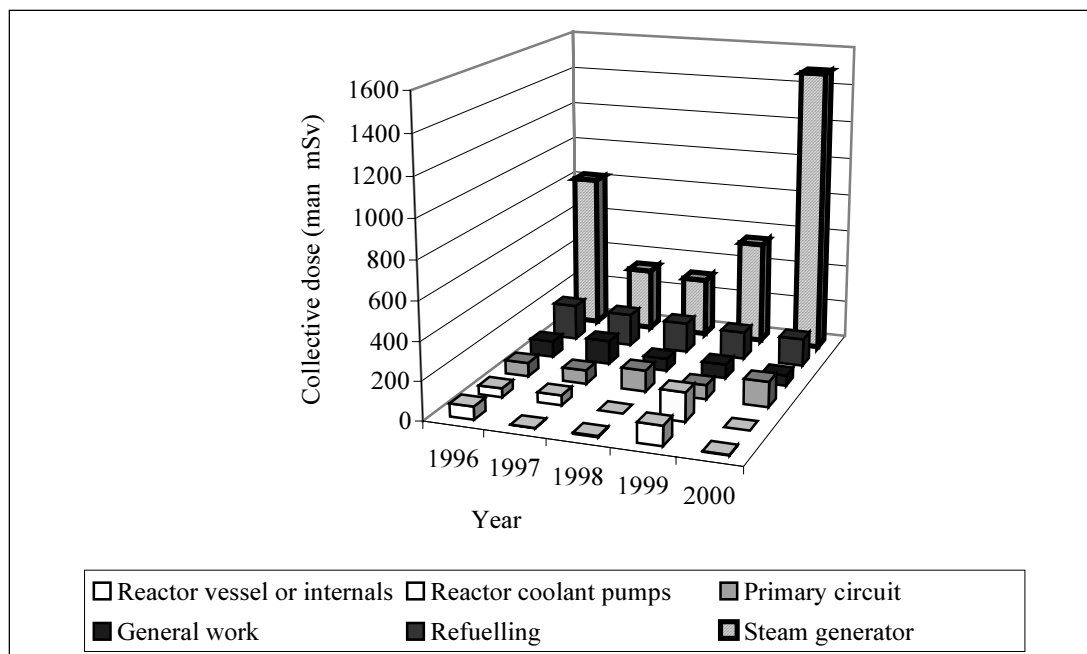


Figure 3 shows the relationship between partial collective doses and the different jobs during outage period from 1996 to 2000. Only those types of work are analysed where the annual collective dose was at least once higher than 100 man mSv in that period. The work related to the maintenance, control and replacement of both steam generators resulted into highest partial collective doses. The next highest dose values belong to the refuelling but in contrast to the work related to steam generators do not show any big variations. It can be seen from the figure that in the year 1999 the works at the reactor coolant pumps and the reactor vessel resulted also in a significant increase of the annual collective dose.

Figure 3. The annual collective doses in the Krško NPP from 1999 to 2000 related to different tasks during outages are given. Only those types of the work are given where the annual collective dose was at least once higher than 100 man·mSv in that period.



Conclusions

The control of occupational doses in the NPP is an exciting issue due to diversity of radiation sources as well as diversity of jobs which are performed in the NPP. Besides primary dose quantities also the subsidiary dosimetric quantities as for example the collective dose can be used in order to follow the implementation of safety standards. However, the use of subsidiary dosimetric quantities has limitations which should be taken into account.

The general trend of the annual collective dose in the Krško NPP in the period from 1990 to 2000 was mainly driven by the maintenance works at the ageing steam generators and by the replacement of both steam generators in the year 2000. In the future, a substantial dose reduction can be expected. This expectation relies already on the dose data from the year 2001. However the dose reduction plan related to the specific sources and specific jobs still remains a challenging issue in the modernised and upgraded NPP.

References

- [1] Nuclear Energy Agency, Occupational Exposures at Nuclear Power Plants, Tenth Annual Report of the ISOE programme, 2000, OECD (2001).
- [2] International Commission on Radiological Protection, 1990 Recommendations of the International Commission on Radiological Protection, ICRP Publication 60, Pergamon Press, 1991.
- [3] The Council of the European Union, Council Directive of 13 May 1996 Laying down Basic Safety Standards for the Protection of the Health of Workers and the General Public against the Danger Arising from Ionising Radiation, Council Directive 96/29/Euratom, Official Journal European Communities L 349, 21-25 (1990).
- [4] Janžekovič, H., Breznik., B., Jovanovič, P., Rojc, J., Štuhec, M., Vaupotič, J., Zdešar, U., Dose Data Analysis of the Occupationally Exposed Workers in Slovenia in Year 2000, Book of Abstracts, IRPA Central European Congress, Dubrovnik (2001), in press.
- [5] M Černilogar Radež, M. Križman, Computerised Registration System of Occupational Exposure in Nuclear Fuel Cycle in Slovenia, Proceedings of the IRPA Central European Congress, Dubrovnik (2001), in press.
- [6] International Atomic Energy Agency, Safety Series No. 115, International Basic Safety Standards for Protection against Ionising Radiation and for the Safety of Radiation Sources, IAEA, 1996.
- [7] The Council of the European Union, Council Directive of 4 December 1990 on the Operational Protection of Outside Workers Exposed to the Risk of Ionising Radiation during their Activities in Controlled Areas, Council Directive 90/641/Euratom, Official Journal of European Communities L 159, 1-114 (1996).

PRACTICAL IMPLEMENTATION OF THE 96/29 EURATOM DIRECTIVE TO THE RADIATION PROTECTION PROGRAMMES OF SPANISH NUCLEAR POWER PLANTS

**O. Guzmán, T. Labarta, J.J. Ontesinos,
M.L. Rosales, M.J. Muñoz, I. Amor**
Spanish Nuclear Safety Council (CSN), Spain

Introduction

The Spanish legislation on Licensing of Nuclear and Radioactive Facilities (Royal Decree 1836/99, Art. 20) requires submission and approval of the so-called “Radiation Protection Manual (RPM)” in the licensing process of a Nuclear Power Plant (NPP). The RPM is a document reflecting the practical implementation of the licensee responsibility for radiation protection through the adoption of management structures, policies, plans, training, procedures and other measures developed and implemented to achieve continuing compliance with the legislation in force and to apply the ALARA principle.

In Spain, the Regulation on Sanitary Protection Against Ionising Radiations (Royal Decree 783/01) came into force on 6 July 2001 and:

- Implement the majority of the Basic Safety Standards Directive 96/29/Euratom (BSS Directive) in Spain.
- Replace Regulation on Sanitary Protection Against Ionising Radiations (Royal Decree 53/92) which was made in response to the 1980 BSS Directive 80/386/Euratom (as amended by 84/467/Euratom).

In the past, RPM complied with the replaced R.D 53/92, the structure and content fitting the Nuclear Safety Council Safety Guide 7.6 “Content of the Radiation Protection Manuals in radioactive and nuclear facilities of the fuel cycle”. The necessary revision of the RPM owing to the new RD 783/01 entailed a clear occasion to achieve a higher degree of homogeneity in the scope and content of some issues that could be subject to interpretation.

Seizing the opportunity, the CSN and the Spanish Association of Electricity Utilities (UNESA) set up an ad-hoc working group constituted by the NPP Radiological Protection Department Heads and the Regulatory Occupational Radiological Protection experts. The group began its activities in February 2000 and it concluded in December 2001.

The driving force of the group was to lay the foundations of a generic document based on homogeneous and coherent radiological criteria that could be used by the facilities as a guidance to produce their respective Radiation Protection Manuals. Each individual RPM should eventually be tailored to the particular features of the facility.

As a result of the work carried out a new framework was established, on the one hand new reference values related to the protection of both workers and the public were set up, on the other hand harmonised methods for ensuring compliance with R.D. 783/01 requirements were included, such as:

- Establishment of limiting values and reference levels for every one of the radiological areas in terms of dose rate, airborne contamination and surface contamination.
- Establishment of reference levels for internal and external dosimetry and surface contamination.
- Regulation of access and permanence conditions for visits in radiological areas.
- New values for ALIs (Annual Limits of Intake) and DAC (Derived Air Concentration) resulting from 20 mSv of annual dose and a working year of 2 000 hours.
- Reuse criteria for clothing and protective equipment

In the next sections the most remarkable features either because of their novelty or because of the uniformity achieved among all facilities are presented:

Classification of areas

The next table presents the accepted values in terms of Dose Rate (DR), Airborne Contamination (AC) and Surface contamination (SC):

	Free access	Supervised	Controlled			
			Free permanence	Limited permanence	Restricted permanence	Prohibited access
DR	< 0.5 $\mu\text{Sv/h}$ (1)	< 3 $\mu\text{Sv/h}$	< 25 $\mu\text{Sv/h}$ and	< 1 mSv/h and	< 100 mSv/h and	>100 mSv/h or
AC		<0.4 Bq/cm ² β/γ < 0.04 Bq/cm ² α Averaged on 300 cm ²	< 4 Bq/ cm ² β/γ < 0.4 Bq/ cm ² α Averaged on 300 cm ² and	< 40 Bq/ cm ² β/γ < 4 Bq/ cm ² α Averaged on 300 cm ² and	< 400 Bq/ cm ² β/γ < 40 Bq/ cm ² α Averaged on 300 cm ² and	> 400 Bq/ cm ² β/γ > 40 Bq/ cm ² α Averaged on 300 cm ² or
SC			< 0.1 DAC	< 1 DAC	< 10 DAC	> 10 DAC

(1) Exceptionally, in those areas were restriction of access in not efficient higher dose rates can be allowed provided that the dose rate is always lower than 2.5 $\mu\text{Sv/h}$. Nevertheless, these areas will be subjected to an administrative radiological control.

Reference levels for the radiological surveillance of areas

The following reference levels are consistent with the established classification of areas:

RL: Recording level

IL/IL: Investigation/intervention level

		Controlled area	Supervised area	Free access area
DR	RL	3 μ Sv	0.5 μ Sv	0.5 μ Sv
	IL/IL	The highest dose rate value which defines the area		
SC	RL	4 Bq/cm ² β/γ and 0.4 Bq/cm ² α	0.4 Bq/cm ² β/γ and 0.04 Bq/cm ² α	0.4 Bq/cm ² β/γ and 0.04 Bq/cm ² α
	IL/IL	The highest surface contamination value which defines the area		
AC	RL	< MDA (< 0.05 DAC for β/γ radiation)		
	IL/IL	The highest airborne contamination value which defines the area		

Reference levels for internal and external dosimetry and skin contamination

	External dosimetry		Internal dosimetry	Surface contamination (1)
	H _p (10)*	H _s (0.07)**		Skin dose due to skin contamination
RL	0.1 mSv/month	0.1 mSv/month (A) and 0.1 mSv/month (B)	1 mSv/year (2)	4 mSv/month averaged on 1 cm ²
Inv. L	10 mSv/year (A) & 3 mSv/year (B).	50 mSv/month (A) and 15 mSv/month (B)	1 mSv/year	50 mSv/month
Int. L	18 mSv/year & 90 mSv in a consecutive 5-y period (A) y 5 mSv/year (B).	450 mSv/year (A) and 135 mSv/year (B)	5 mSv/year	450 mSv/year

* Individual dose equivalent, penetrating, H_p(d)

** Individual dose equivalent, superficial, H_s(d)

1. The reference level set up for skin contamination averaged on 100 cm² is 4 Bq/cm² for β - γ emitters. In case α emitters have to be measured, the reference value would be 0.4 Bq/cm².
2. For those radioisotopes that, by virtue of their physical-biological characteristics, this value it is not compatible with the Minimum Detectable Activity of the measurement technique, the approach established for the recording of dose will be followed whenever activities higher than the MDA are measured.

Non exposed workers

A new figure has been introduced, non exposed workers, not considered in the R.D. 783/01. This figure, included in the group “Members of the Public”, covers those cases of certain not large groups of non exposed workers, who may enter low dose rate and contamination areas in the free permanence controlled areas (green).

For these workers dose is limited for every working period in the power station so as to guarantee that the incurred annual dose does not exceed 1 mSv. Thus a 40 μ Sv dose constraint is set for every working period at the NPP. If the dose constraint is exceeded, then it must be justified.

Moreover, it has been required to keep a register of every visit or non exposed workers who enter the controlled area.

ALIs and DACs

The RPMs of the Spanish NPP have included as an annex the new Annual Limits of Intake (ALIs) and Derived Air Concentration (DACs) values. The calculations have been carried out by the electrical sector using Directive 96/29 dose coefficients and are based on a committed effective dose of 20 mSv and a 2000-hours working year.

Protective clothing and equipment

For the performance of works with risk of contamination, access is not allowed to radiological areas wearing personal clothes, with the exception of underclothing.

Criteria concerning surface contamination levels acceptable for the reuse of personal clothing and protective equipment have been standardised. Such levels, averaged on 100 cm² are presented in the next table:

Emitters	Clothes in contact with the skin		Clothes worn above the previous ones		Reuse of respiratory equipment	
	β - γ and low toxicity α	Remainder α emitters	β - γ and low toxicity α .	Remainder α emitters	β - γ and low toxicity α	Remainder α emitters
fixed contamination	< 4 Bq/cm ²	< 0.4 Bq/cm	< 40 Bq/cm ²	< 4 Bq/cm ²	< 4 Bq/cm ²	< 0.4 Bq/cm ²

Calibration, operation tests and maintenance of systems and equipment

A new section has been introduced where the general rule concerning calibration frequency for the different measurement systems and equipment is described. The rule allows modifications that can be justified based either on manufacturer recommendations or operational experience of the equipment itself. Calibration frequency will agree with the Spanish Norm UNE-EN-300012-which deals with metrological quality assurance.

PRESENT AND FUTURE ASPECTS OF HARMONISATION IN OFFICIAL AND OPERATIONAL PERSONAL DOSIMETRY

S. Hübner, W. Wahl
GSF Research Centre, Germany

Abstract

New international standards in the field of personal dosimetry were set up in the recent years. Resulting requirements appointed in the national Radiation Protection Ordinance actuated the discussion about necessary developments and the future structure of the radiation protection surveillance.

The latest evolution of Electronic Personal Dosimeters (EPD) reaches a raised technical level fulfilling the new EC recommendations and the requirements for practical radiation protection in case of external exposure to ionising radiation. The Physikalisch-Technische Bundesanstalt (PTB) in Braunschweig, Germany assigned the first type-approvals according to the new requirements for modern EPDs. In some European countries EPDs can already be applied as official dosimeter under certain conditions.

After characterising the present structure of radiation protection surveillance in Germany, proposals for the future developments in official Personal Dose Monitoring System will be discussed in this paper, taking into account the different demands, interests and aims of all the different parties hereto, namely the operators of the installations with their workers kept under surveillance, the government and competent authorities, the manufacturers of modern dosimetric systems and last but not least the official monitoring centres.

Introduction

The new EURATOM “basic-safety-standards” (Council Directive 96/29/ EURATOM) [1] for the protection of the health of workers and the general public against the dangers arising from ionising radiation were to be put into action in the EU member countries. Based on the ICRP-60 recommendations [2], this directive introduced new radiological requirements to the official dosimetry. With the commencement of the new national radiological protection ordinance [3], all provided official dosimeters have to fulfil the new radiological requirements within the national transition period, in particular lower dose limits for the population and occupational radiation exposed workers and dose constraints as well as new operational quantities “deep dose” Hp(10) and “shallow dose” Hp(0,07) for personal monitoring.

In addition, the new IEC 61526 standard [4] in Radiation Protection Instrumentation “Measurement of personal dose equivalents Hp(10) and Hp(0,07) for X- and γ -rays, beta and neutron

radiation” especially for “Direct reading personal dose equivalent and/or dose equivalent rate dosimeters” suggests to check EPDs for their use as legally approved dosimeters.

In most countries, official dosimetry is carried out using passive dosimeters. But recently substantial efforts have been done in several European countries promising an introduction of EPDs in the official dosimetry in the near future. In some installations in France EPDs can already be applied for official dosimetry on condition of the use of a second dosimetric system whereas in Switzerland it is possible to use EPDs as exclusive legal dosimeters upon request. In UK, EPDs already have replaced passive detection systems as legally approved dosimeters in some specific workplaces [5].

In Germany, EPDs are up to now used only for operational but not for legal personal dosimetry, which is performed using passive dosimeters such as film badges or solid-state detectors. However, recent improvements of performance and reliability of EPDs, as well as their interesting technical features and advantages over passive radiation detection systems inevitably lead to substantial discussions about the acceptance of EPDs for official personal dosimetry.

To find possibilities to apply EPDs for official personal dose monitoring in Germany, a pilot project is currently ongoing. With the collaboration of all involved decision-makers proposals for the inclusion of EPDs in the official dose monitoring are elaborated, reflecting the respective specific interests and demands. Even though the framework of state law and related guidelines does not exclude official use of EPDs, as an intermediate result of this project it is pointed up that binding interpretation of the wording of the laws is mandatory.

Present situation of the German personal dose monitoring

Official monitoring service

Based on the national Radiation Protection and X-ray Ordinance, in Germany the legal personal dosimetry is to be carried out by six accredited monitoring services, meeting the Guideline “Requirements for Personal monitoring services”. Appointed by the competent authority according to the state law, these monitoring stations with different regional competence have to provide, distribute and evaluate suitable dosimeters for legal personal dosimetry. Whole-body doses are quantified by measuring the personal dose equivalents $H_p(10)$ using personal dosimeters worn on the trunk of the body. To be determined as legal by the competent authority, the personal dosimeters have to have a PTB-type approval. Different well-established passive dosimeters are presently used for the official personal dosimetry.

Film dosimeter

The surveillance of occupational radiation exposed people is mainly carried out using different film badges. At present more than 320.000 film dosimeters are monthly distributed and evaluated by the official measuring centres in Germany. New developments enable the measurement of the effective dose $H_p(10)$ with a maximum deviation smaller than the permissible overall measuring uncertainty using gliding shadow badges.

Even though the uncertainty of other systems might be slightly smaller, film badges are an excellent choice because they are low priced and intend to provide important information on the conditions of exposure to ionising radiation.

Solid-state dosimeter

Besides the film badges, about 32 000 glass-dosimeters are in use to measure Personal Doses due to photon radiation. At last, thermoluminescence detectors are applied in the official personal dosimetry due to neutron radiation using ALBEDO dosimeters (about 5 000) as well as in the official extremity dosimetry using different types of finger ring dosimeters (about 12 500).

Operational personal dosimetry

Normally, operational personal dosimeters are distributed to the worker kept under surveillance and evaluated by the operator of the plant or installation.

In plants and other installations, operational dosimeters can be stipulated by the competent administrative authority in addition to and independent from the respective legal dosimeters. Recurring calibrations or control measurements are obligatory for those ordered secondary dosimeters. Particularly in NPPs, direct-reading operational dosimeters must be applied. For this purpose EPDs are in use.

Eligible requests for the innovation of personal dose monitoring

Within the scope of the implementation of new standards and recommendations, substantial demands and requests reflecting the state of the art were raised for modern radiation protection.

Besides the adherence of international recommendations, the focus of government and authorities is the surveillance of the compliance of dose limits and to assure the official control of the national physical radiation protection.

Especially in installations with stipulated additional direct reading dosimeters the current situation of the official dose monitoring using passive detectors turned out to be unsatisfying. Because of possible differences between the promptly available operational doses using EPDs and the official doses provided by evaluating e.g. film dosimeters not until the following month, the main interest of the operators of NPPs is to legalise EPDs. Additionally, modern electronical dosimetry systems provide more reliably doses. Thus the use of EPDs would implicate better and cheaper radiation protection in NPPs.

The majority of international and national ordinances and guidelines traditionally are passed based on the supposition that official dosimetry can only be performed using passive dosimeters. Recent developments and improvements of active dosimetric systems were not taking into consideration. Thus, existing recommendations cannot be applied to modern EPDs in an unequivocal way. In order to develop and produce most modern and suitable dosimeters fast and economically, manufacturers shall be entitled to demand explicit and reliable specifications. The licensed monitoring services are officially mandated to provide suitable dosimeters and to ensure the routinely evaluation of the personal doses. They have to reconcile securing of the best possible protection of occupational radiation exposed workers and cost-covering running of the measuring service. In order to meet their official mandate now and in the future, monitoring services should use state of the art technology and, forward looking, participate in technical progress.

Electronic personal dosimeter

Currently, electronic dosimeters are predominantly used as a secondary dosimeter. However, continuous improvements in technique and data security led to modern EPDs, meeting the requirements for Personal Dosimeters according to the national Radiation Protection Ordinance and related guide-lines [6]. Several models of EPDs are already type tested as personal dosimeter by the PTB, and consequently showing their possibilities for official personal dosimetry. This potential is underlined by extensive experiences with EPDs gained in NPPs as well as in the NATO for many years.

In contrast to all passive detection systems, direct reading EPDs with alarm devices lead to dose reduction for monitored people and thus enable active and practical radiation protection. Information about dose and/or dose rate is available in real time. Acoustic personal warning signals allow immediate reaction in case of a sudden increase of the radiation exposure.

Compared to current available passive dosimetric systems, the occurrence of erroneous handling (not wearing in the control areas, exchanged assignment, etc.) or manipulation of EPDs appears to be negligible. Provided that suitable radiological properties are warranted by the type-approval, EPD systems would consequently provide comparative reliable personal doses.

Personal dose monitoring in the future

In the future, personal dose monitoring will undergo technical and structural changes, due to new legislation, harmonisation and liberalisation within the European Community as well as technological progress. The importance of innovation in personal dosimetry is underlined by founding the a EURADOS working group “Harmonisation of Individual Monitoring”, funded by the European Commission in the fifth framework program and other participating institutes.

Next to established passive detectors modern active systems will increasingly play an important role in official personal dose monitoring. This innovation will be accompanied with new techniques of dose evaluation and data processing applied in the radiation protection surveillance. In NPPs and other installations, where direct reading operational dosimeters are required or reasonable, the application of official EPDs should be demanded. Ongoing developments of electronic dosimeters (EDs) will be pushed to meet also the request of modern and accurate systems measuring neutron or neutron/photon radiation fields. Additionally, EPDs providing detailed information about exposure condition, as it is possible evaluating film dosimeters, are in the stage of development.

For the future, wireless remote monitoring of people working outside or with open activities means a further evolution in the field of personal dosimetry. Specified workers could be kept under surveillance applying real-time dosimetry systems in online monitoring stations for both, photons and neutrons.

In addition, it is considered important to further pursue harmonisation along these lines by forming.

Conclusions

An increasing number of institutes as well as individuals kept under surveillance according the Radiation Protection and X-ray Ordinance demand a more transparent and reliable official personal dosimetry. As a matter of principle modern, EPDs are able to meet these requests.

In order to realise legal personal dosimetry using EPDs binding specifications and requirements are inevitable. Therefore, the existing ordinances and guidelines are to be interpreted in terms of legally used EPDs or upgraded in the case of need.

References

- [1] European Commission. Council Directive 90/641/EURATOM of 13 May 1996, Official Journal of the European Communities L 159, Vol. 39 (29 June 1996).
- [2] International Commission on Radiological Protection. 1990 Recommendations of the International Commission on Radiological Protection. ICRP Publication 60. Annals of the ICRP Vol **21** n° 1-3, Pergamon Press, Oxford (1991).
- [3] Bundesgesetzblatt Jahrgang 2001 Teil 1 Nr. 38, *Verordnung für die Umsetzung vom EURATOM-Richtlinien zum Strahlenschutz*, Bonn, 1713-1847, 26 July 2001.
- [4] International Electrotechnical Commission, Radiation protection instruments – Measurement of personal dose equivalents $H_p(10)$ and $H_p(0.07)$ for X, gamma and beta radiations – Direct reading personal dose equivalent and/or dose equivalent rate dosimeters. International Standard IEC 61526, (IEC, Geneva) (1998).
- [5] Clarke P.W. and Weeks A.R., Implementation of an electronic personal dosimetry system (EDP) at Oldbury-on-Severn power station, *J.Radiol.Prot.* **21** (2001) 45-55.
- [6] Ortega X., Ginjaume M., Hernández A., Villanueva I. and Amor I. *The outlook for the application of electronic dosimeters as legal dosimetry* *Radiat. Prot.Dosim.* **94**(2001) 87-91.

**THE EXERCISE OF THE ALARA PRINCIPLE
AT THE INSTALLATION OF THE PRIMARY NEUTRON SOURCES
INTO THE FUEL ASSEMBLIES AT THE TEMELÍN NPP**

J. Koc

Temelín NPP, Czech republic

K. Petrová

State Office for Nuclear Safety, Czech republic

The Temelín NPP consists of two units VVER 1000 type V320, each with a rated thermal power of 3 000 MW. The original Soviet design has been modernised in the early nineties and the original instrumentation and control system, the diagnostics as well as the complete radiation control system have been replaced. The modifications touched also the supply of the nuclear fuel. Currently, both Units are in the stage of commissioning, Unit 1 in the power ascension testing stage at a power of 100 % N_{rated} , Unit 2 is in the stage of zero and low power testing.

Installation of the primary neutron sources into the fuel assemblies

A unique working operation not ever repeated later during the unit operation is connected with the process of the reactor fuel loading preparation, namely the process of the primary neutron sources (PNZ) installation into two determinate fuel assemblies. Those fuel assemblies are located in a cylindrical container with another fuel assemblies containing the nuclear fuel. The reactor core of the Temelín NPP VVER 1000 unit incorporates altogether 163 fuel assemblies, comprising 312 fuel rods each with fuel enrichment max. 5% ^{235}U (the first loading has an average core enrichment of 2,22 %). The complete fuel loading is 92 t and the cycle of the fuel replacement is four years.

Prior to commencement of the fuel loading into the reactor vessel, it is necessary to install into two concrete fuel assemblies one primary neutron source into each consisting of a long supporting metal wire with a capsule containing ^{252}Cf in the form of a wire of Pd-Cf₂O₃. The fuel supplier in accordance with relevant working procedures performs these working activities. From the experience acquired at the course of the installation of PNZ designated for the Unit 1, which was carried out in the first half of 2000 it follows that the future unit operator must be actively involved both into the preparation and into the implementation of the activities related to the installation of the primary neutron sources themselves. For one thing experience acquired during the works made by them supplier's workers and particularly an out of proportion high value of the collective effective dose (KED) and the individual effective dose (IED), that received the workers during the installation of the primary neutron sources led to such conclusions.

The status of the alara principle implementation at the Temelín NPP prior to the fuel loading into Unit 1

At the time of the primary neutron sources installation for Unit 1, the utility control document regulating the requirements for the ALARA principle application at ensuring activities involving ionization radiation sources has been already in the process of the development. Also, an ad-hoc instruction of the plant director has been already issued for the organisational and technical provision of the ALARA principle implementation at the Temelín NPP, on the other hand however, conditions have not yet been set up for its full accomplishment in the practice.

Series of activities related to the process of the PNZ installation into the fuel assemblies having been performed, as a consequence of such situation, more or less spontaneously and ad hoc without having performed due analyses of potential impacts of the operations course on the radiation situation at the workplace. The result of above described situation was the necessity of an flexible temporal closure of a part of the building with the declared and defined temporal controlled zone, where the fuel assemblies' container with installed PNZ was located till the fabrication of a sufficient additional shielding of the container where the container has been located with the fuel assemblies before the fuel loading into the reactor core. The shielding container in the form of a cylinder is comprised of a vessel with double walls 30 cm of each other. The created ring is filled with boric acid solution with a concentration of 50 g/kg.

The lack of enough experience and not completely sufficient preparedness of the workers involved in the assembly of PNZ and the insufficient implementation of the ALARA principle resulted in a personal effective dose of neutrons of 6,73 mSv of one worker during the mounting of the primary neutron sources into the fuel assemblies on May 18, 2000. Subsequently, on June 2, 2000 during the installation of the container of the fuel assemblies containing primary neutron sources into the shielding cylinder, another two workers received individual effective doses of neutrons 1,55 mSv and 1,43 mSv. Outside the temporary controlled zone, where all manipulations have been performed and where the access was restricted only for a limited group of workers, the radiation protection staff did not measure any increase of the equivalent dose rate level (PED) from neutrons due to performed working operations above the level of the natural background. This was verified even based on the official measurement made by workers of the Czech Metrological Institute.

The status of the ALARA principle implementation at the Temelín NPP prior to the fuel loading into Unit 2

During 2000 and 2001, a marked positive shift occurred at the Temelín NPP in the field of the ALARA principle implementation. It came to restructurization and to personal replenishment of the department of the Radiation Protection Control and its staff got involved into the realization of the project TC RER/9/063 Occupational Radiation Protection at Nuclear Power Plants. By means of participation at the 5th and 6th workshop of the project they acquired both generally valid information in the area of the issue of ALARA principle implementation at the NPPs and, in particular, established personal contacts to the workers of other NPPs of the VVER and RBMK types.

In 2001, the control document Quality Assurance Procedure 27.08.01 Radiation Protection Control has been issued at the Temelín NPP, which defines the statute of the newly established ALARA Commission. The control document defines among others the so-called works with an increased radiation risk, to the realisation of which is necessary the development of the Program of the Radiation Protection Assurance (PZRO). PZRO precisely describes and determines the radiation protection measures leading to assurance of the radiation doses optimization at performing activities

connected with radiation exposure risks. All departments involved in the planned activity co-operate on the PZRO development. An authorized expert of the radiation protection performs the final approval of the PZRO after an analysis of the complete program.

In the same time, a workplace has been established and provided with appropriate computer and audiovisual instrumentation at the Temelín NPP that set up conditions for providing for training, consulting and pre-work meetings of the workers in the frame of the so-called PRE JOB TRAINING.

Installation of the primary neutron sources into the fuel assemblies for Unit 2

Prior to commencing the assembly of the PNZ into the fuel assemblies for Unit 2, it has been already proceeded in accordance with in advance established rules introduced in the frame of the ALARA principle implementation at the Temelín NPP. Detailed procedures of manipulations have been developed, assessments of individual effective doses from neutrons for individual workers involved in the manipulations with PNZ have been made and the organisational measures to provide for safe development of the complete operation have been taken. All measures have been included into the Radiation Protection Assurance Program No. 00 00 11/2001 (see Annex 1), that was binding for all workers participating on the manipulations with the PNZ.

One day before the commencement of the assembly itself, training of all persons involved in the manipulations in the material contents of the radiation protection program approved to the activity. In the same time, simulated drill has been performed of working operations related to the assembly of PNZ into the fuel assemblies and preparatory works for loading of the fuel assemblies' container into the shielding container. The second drill of the PNZ assembly has been performed one hour before commencing the manipulations themselves. Both drills have shown that all working operations connected with the primary neutron sources installations into the fuel assembly may be performed in a significantly shorter time period then assumed by the developed Radiation protection assurance program. For conservative reasons, however, it has been still adhered to the original time schedule from the point of view of forecast doses.

Evaluation of the primary neutron sources installation for Unit 2 in accordance with the developed PZRO

All manipulations planned in accordance with the PZRO took place precisely in compliance with the working procedures. Duration of all activities has been substantially shorter then originally assumed, in particular due to performed training and drills. The real PED of neutrons at the workplace has been also lower than the original conservative assumption, which was based on calculations and results of measurements carried out at manipulations for Unit 1. The doses received from neutrons have been therefore due to shorter times and lower neutron PED lower than assumed, too.

All involved persons have been provided with neutron, film and electronic personal dosimeters. The radiation protection department staff by means of portable radiometers has continually monitored the radiation situation in the space of manipulations. The received doses have been for all participating workers below the sensitivity lower limit of the neutron and film dosimeters. The maximum IED of the gamma radiation registered by electronic personal dosimeters of the EPD1 type reached the value of 9 μSv .

During the manipulations, all involved workers have been in an environment wit a maximum PED of neutrons 400 $\mu\text{Sv/h}$ and of gamma radiation 6,5 $\mu\text{Sv/h}$. The stays of individual workers in the

maximum field of neutron PED did not exceed 90 seconds. Annex 1: Form of the Radiation Protection Assurance Programme.

Radiation protection assurance programme

For the work at increased radiation risk long-term
PZRO No. 00 00 11/2001

The coloured fields to be filled by the Program developer	
Title of the working activity:	Assembly of the primary neutron sources and relocation of the fuel assembly container with primary neutron sources into the water shielding
Technological procedure (title, developer, No.):	VVantage-6 Primary Source Site Assembly and Installation Procedure
Place of the working activity (room/equipment):	BAPP-01/134a
Radiation work permit No.:	
Date of the work:	October 31, 2001
Work leader (name, dept. ETE/company):	Ernst Daniel, dept. 4521
The assumed time of the works (number of workers, number of hours necessary to perform the work):	Ref. to text

Description of the working operations, list of rooms, work duration in individual rooms, numbers of workers:
1. Live assembly of the primary neutron sources into the fuel assembly containers in SČP (fresh fuel storage) 2. Transfer of the container for the fuel assemblies with mounted primary neutron sources into the cylindrical water shielding in SČP (room 134a) <i>Annexes</i> Annex 1: Detailed description of activities, times and an doses estimate for both operations Annex 2: Expression of the dependence of neutron PED from both PNZ on the distance Annex 3: Working procedure of the performed activities

Technical and administration measures to ensure the radiation protection:

General measures:

1. Prepare the cylindrical shielding with the boric acid solution with a concentration of 50 g/kg, all necessary aids and tools, to make marks for manipulations prior to the works commencement.
Ensured by: 4521, 4414
2. Close and evacuate the building 801/01 including closing of the access to the pavement along the building 801/01, to the access roads to the entrances to 116a, 130a, 132a and on the roof of the building 801/01 on October 31, 2001 from 06:00 till 16:00 hours.
 - provide for presence of 1-2 workers at the entrance from 801/02 to 801/01
 - provide for presence of 2 workers along the building 801/01Ensured by: 4020
Provide the pavement and access roads closure around the building 801/01 for by the band with the notice „No entry – dangerous invisible radiation,,
Ensured by: 4542, TPRK
3. Perform continual monitoring of neutron PED and of gamma PED for the complete time of manipulations with the PNZ. The TPRK, which will perform the monitoring including recording of the times of individual working operations, will be as close to the involved persons as possible and won't approach to the PNZ at a distance lower than 5 m. In case of a sudden neutron PED or gamma PED increase he will give order to stop the work and to leave to a safe distance. The work continuation may be permitted only after finding and possible removal of the reason of the PED increase or after re-assessment of the occurred radiation situation by a radiation protection expert.
Ensured by: 4542, TPRK, radiation protection specialist
4. Measure the neutron PED at the outer side of the BAPP-01 (auxiliary) building in the middle of the walls of the room 132a after removal of the PNZ from the transport container, during the assembly of PNZ into the fuel assembly and during its transport into the cylindrical shielding. Provide for presence of a ČMI (Czech Metrological Institute) worker, assign him the measurement location. Perform neutron PED monitoring from the outer side of the BAPP-01 building in the following order: wall 113a, wall 123a, wall 132a.
In case of neutron PED increase outside of the BAPP-01 building above 2,5 $\mu\text{Sv/h}$, pass TPRK1 this information to TPRK2 inside BAPP-01. Then, TPRK1 will perform the check of the radiation situation on the border of the defined space and, if necessary, will extend the space so that the neutron PED is not higher than 2,5 $\mu\text{Sv/h}$ on the border of the space. The instant of PNZ removal and end of the manipulations will announce TPRK2 inside BAPP-01 by means of a transmitter.
Ensured by: 4542, TPRK2 inside BAPP-01 announcement of the start and completion of the manipulations, TPRK1 outside BAPP-01 assignment of the places for measurements, the VSRK on duty by means of a transmitter.
5. All workers present at the manipulations with PNZ will keep as far as possible away from the PNZ. If they won't directly perform the PNZ assembly and transport, they won't approach nearer than 5m.
6. Record the course of manipulations with the PNZ by means of a video camera. Ensured by: 4541.
7. Provide for the co-ordination of the activities. Ensured by: 4541.

8. Adhere to the work conditions with the electronic dosimeter and know the procedures at the increase signalling. Adhere to the procedure described in the Instruction for the electronic dosimeter use, which is kept on the bulletin board Radiation Protection. Ensured by: All workers

Equipment by special dosimeters:

In addition to standard electronic and film dosimeters, equip all workers present at the manipulations with PNZ also with neutron dosimeter and TLD dosimeter.

Set up the alarm levels for personal electronic dosimeters:

- low alert: Hp(10) 800 μ Sv, PED 800 μ Sv/h
- high alert: Hp(10) 1 mSv, PED 1 mSv/h

Ensured by: 4544

Use of supplemental OOP (personal protection aids) in accordance with the measured radiation situation:

None

Changes, modifications and measures (description, the modifications have been entered by - date, name, first name, signature):

None.

List of workers' names providing for manipulations:

4414

Ing. Jiří Bigas, Miroslav Kolář, Dušan Sadílek, Evžen Sznepka, Karel Vidlák

4541

Jaroslav Hak, Mgr. Jiří Vokálek

4521

Ing. Daniel Ernst

WEC

John Wood, John Meskanick

4542

František Pokorný, Daniel Janovský

Maximum IED (mSv):	1,33	Planned KED (mSv):	7,94
Developed by:			
Name:	Department:	Date:	Signature:
Vokálek Jiří	4541	October 25, 2001	
Endorsed by:			
Name:	Department/company:	Date:	Signature:
Hak Jaroslav	4541	October 25, 2001	
Approved by:			
Name:	Department:	Date:	Signature:
Ing. Koc Josef	4540	October 26, 2001	

THE IMPLEMENTATION OF THE RADIOLOGICAL SAFETY STANDARDS IN UKRAINE

T. Lisova

Ministry of Fuel and Energy, Ukraine

V. Novikov

NNEGC "Energoatom", Ukraine

Introduction

Standard acts directed at the increase of nuclear enterprise personnel and public protection from ionising radiation were adopted in Ukraine during the recent 5-7 years. While developing these laws the latest accomplishments of the world nuclear community in the area of radiation protection generalised in ICRP (46, 60, 64, 76, 77, 81) and IAEA publications were used.

The main content of the new Radiation Safety Standard of Ukraine (RSSU-97) and the Law of Ukraine "Protection of Man from Acting Ionizing Radiation About"

The Law of Ukraine "Protection of Man from Acting Ionizing Radiation About" was adopted on February 24, 1998. This Law defines the right of man to be protected from ionizing radiation exceeding established limits. This right is realized by means of integrated measures to protect man from ionizing radiation exceeding established limits, reimbursement for dose limits exceeding and detriment caused by ionizing radiation impact reimbursement.

Main exposure dose limits are defined:

- Public exposure shall not exceed 1 mSv of the annual effective dose.
- Newly commissioned nuclear facility occupational exposure shall not exceed 20 mSv of the annual effective dose, assuming that the dose can be increased up to 50 mSv on condition that the average annual exposure dose during 5 years does not exceed 20 mSv.
- Operating nuclear facility occupational exposure shall not exceed 50 mSv per 12 months in succession, with gradual decreasing of radiation dose limits to 20 mSv per year during the transition period.

The time of such transition period is defined by state regulatory bodies.

According to this Law protection of man from radionuclides contained in construction materials, foodstuffs, drinking water is ensured. Protection of man during medical practice is ensured.

New radiation safety standard of Ukraine RSSU-97 entered into force 01.01.98. Supplement to it RSSU-97/D2000 entered into force in the middle of 2000. This standard introduces the concept of

potential exposure to ionizing radiation and regulates probability of criticality events connected with exposure. It also defines three basic principles of radiation protection with reference to potential exposure to ionizing radiation.

- Justification Principle – practical activity which can lead to exposure to ionizing radiation shall not be implemented if the benefit for people exposed and for the society in general does not exceed the harm from this activity now and in the future in connection with the potential occurrence of criticality event.
- Non-Exceeding Principle – all types of practical activity falling under sanitary surveillance shall not lead to exceeding of the dose values and probability of potential exposure to ionizing radiation regulated by this document.
- Optimisation Principle (ALARA) – criticality event probability and potential exposure dose as well as the number of persons that could be impacted by such sources shall be as low as reasonably achievable taking into account economic and societal considerations.

These principles are utilized at practical activity planning stage.

“Program of transition of nuclear power enterprises of Ukraine to operation meeting the requirements of RSSU-97” adoption and implementation

In order to take up new dose limits the **“Program of transition of nuclear power enterprises of Ukraine to operation meeting the requirements of RSSU-97”** was developed by the Ministry of Energy of Ukraine and the State Scientific and Engineering Center of Control and Emergency Response Systems and approved by the regulatory body.

Applicability of the Program.

- Definition of radiation safety condition conformity to RSSU-97 requirements.
- Definition of the new standard general transition activity directions.
- Transitive period duration establishment.

The program defines the main directions of the activity:

- Revision of standards (instructions, radiation control regulations) at the plant level.
- Measuring procedures revision and development.
- Occupational exposure reduction measures development.
- Revision of plant radiation monitoring systems including new equipment installation.

Transition period duration is established for the nuclear enterprise or facility where operation is performed proceeding from organisational evolutions terms while taking up new standards.

The program establishes the duration of the first stage for nuclear enterprises for Ukraine’s NPPs-5 years starting January 1, 1998.

The following organisational evolutions have been planned and implemented in the framework of the program:

1. Training of NPP personnel in RSSU general provisions and their differences from the previous RS- 72/87 was organized.
2. NNEGC “EnergoAtom” Radiation Protection Council was created.
3. NPP dose limits of occupational exposure projection groups (ALARA) were organized and Provisions for ALARA group performance were adopted.
4. Transition to monthly measuring of the occupational integral individual doses; women under 45 and critical group personnel (where exposure dose higher than 10 mSv/year is possible).
5. Dose limits exposure analysis for the personnel performing radiation dangerous activities is performed by ALARA groups; collective dose projection report is issued before the planned outage and outage performance report is issued afterwards.
6. The list of radiation dangerous activities is worked out; before the planned outage organisational evolutions to decrease the exposure are developed.
7. Planned action implementation control is performed during the planned outage.
8. Radiation monitoring of permanent and temporary residence of personnel into stringent operating condition zone was performed.
9. Dose limits of NPP personnel occupational exposure projection and analysis techniques are being developed.

Lack of electronic dosimeters at NPPs is a still great problem.

It makes radiation monitoring more complicated and decreases its efficiency.

Measures to develop the internal exposure dose calculations according to the results of body counter measurement techniques for NPPs are not yet implemented, such techniques are developed only for “The Shelter” facility.

Current reduction of occupational exposure doses at Ukrainian nuclear power plants as a result of optimization of the radiological protection (ALARA)

The main measure of activity associated with the radiation protection quality is the collective and average doses.

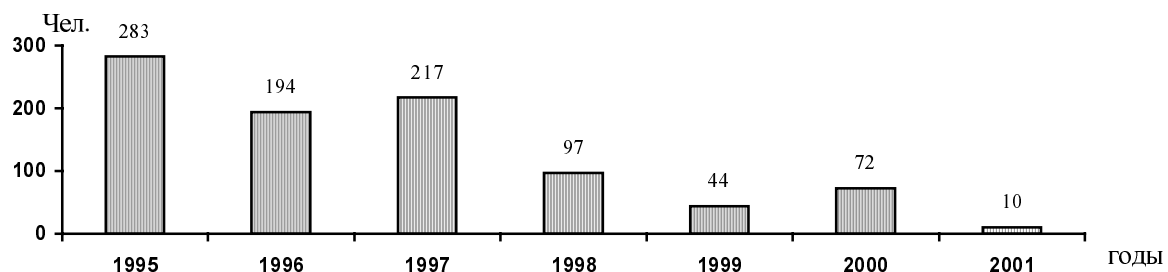
Occupational exposure dose reduction trend is one of the indicators while following the ALARA principles.

Table 1. Individual occupational exposure dose distribution on January 1, 2002 collective and average annual individual doses for 2001

NPP	Number person under control	Number personnel intook dose exposure on January 1, 2002, man,(mSv/year)									Dose exposure for 2001 year, mSv	
		<2	2-5	5-10	10-15	15-20	20-30	30-40	40-50	>50	Collective dose	Average annual individ. dose
Zap.NPP	4789	4204	322	169	70	24	0	0	0	0	4324.1	0.900
Incl. outside personnel	477	443	15	15	3	1	0	0	0	0	247.2	0.520
Rivne NPP	3247	2370	431	166	38	32	10	0	0	0	5173.2	1.590
Incl. outside personnel	415	198	10	7	0	0	0	0	0	0	450.5	1.090
Khmel.NPP	2072	1899	132	39	2	0	0	0	0	0	1377.1	0.665
Incl. outside personnel	420	390	27	3	0	0	0	0	0	0	212.5	0.506
SU NPP	2846	2074	394	217	130	31	0	0	0	0	5872.0	2.063
Incl. outside personnel	517	492	76	45	43	14	0	0	0	0	1273.4	1.901
NNEGC	12954	10547	1279	591	240	87	10	0	0	0	16746.4	1.293

Graph 1 shows NNEGC personnel quantity changes with external individual doses exceeding 20 mSv per year during the recent 7 years.

Graph 1. The number of WWER- reactor NPP personnel having intaken the external dose more than 20 mSv per year during 1995-2001



Graph 1 shows that the number of such personnel has gradually decreased and in 2001 only 10 people intook the doses higher than 20 mSv (approx. 30 mSv) and what's more this decision was

approved by Chief Sanitary Doctor on condition that in 5 years to come the collective dose of these people could not exceed 100 mSv.

Table 2 represents the WWER-reactor NPP personnel and outside personnel collective exposure dose change dynamics for NPP in general and data for the recent 5 years for one unit.

Table 2. WWER-reactor NPP personnel and outside personnel collective exposure dose in 1997-2001

Year/ NPP	1997			1998			1999			2000			2001		
	Num man	Dose, m.Sv	Unit/ m.Sv	Num man	Dose m.Sv	Unit/ m.Sv	Num man	Dose m.Sv	Unit/ m.Sv	Num man	Dose m.Sv	Unit/ m.Sv	Num man	Dose m.Sv	Unit/ m.Sv
Zap. NPP	4561	5.26	0.88	4634	6.74	1.12	4660	3.78	0.63	4779	4.37	0.73	4789	4.32	0.72
Rivne NPP	3745	6.36	2.12	3289	5.78	1.93	3198	4.24	1.41	3589	4.78	1.59	3247	5.17	1.72
Khmel NPP	2061	3.01	3.01	2012	2.02	2.02	2024	1.63	1.63	2070	2.46	2.46	2072	1.38	1.38
SU NPP	2641	12.0	4.00	2721	10.1	3.36	2689	8.61	2.87	2689	8.31	2.77	2846	5.87	1.96

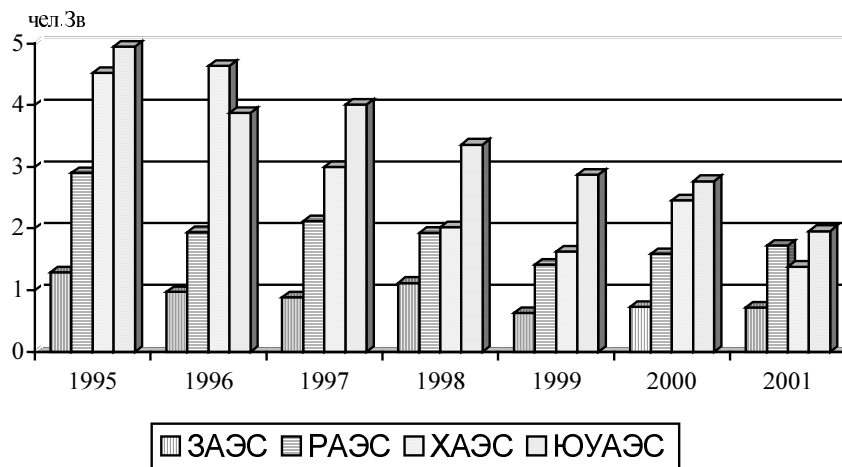
Total collective occupational exposure dose for NPP personnel in 2001 was 16,7 man·Sv, for 2000 it was 17,92 man·Sv ,average collective dose for the unit was 1,38 man·Sv, in 2001 – 1,29 man·Sv.

Sister-unit group data for comparison:

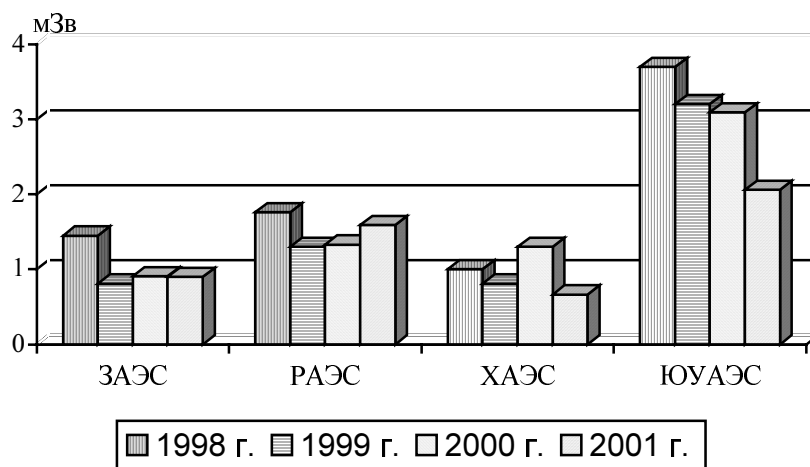
Hungary-0,76 man·Sv/unit, Slovakia-0,81 man·Sv/unit, Finland-1,132 man·Sv/unit, Slovenia-2,59 man·Sv/unit.

The following graphs represent WWER- reactor NPP personnel collective exposure doses and average individual exposure doses for the recent years.

Graph 2. WWER-reactor NPP personnel and outside personnel collective exposure dose in 1997-2001, 1 unit



Graph 3. WWER-reactor NPP personnel average individual exposure dose in 1998-2001



New more stringent limits adoption

In January 2002 a joint meeting of Regulatory bodies (Ministry of Health Protection and State Committee on Nuclear Regulation) and NNEGC “EnergoAtom” took place, where the implementation of the **“Program of transition of nuclear power enterprises of Ukraine to operation meeting the requirements of RSSU-97”** was analyzed. The main objective of the **“Program...”** transition to new NPP personnel occupational exposure dose standards (20 mSv/year), revision of regulations and requirements to meet the new standards, introduction of exposure dose planning during the planned outage etc. **was accomplished.**

RADIATION PASSBOOK AND TRAINING MODEL FOR OUTSIDE WORKERS IN SPAIN

I. Villanueva, A. Hernandez, M.J. Muñoz, I. Amor
Spanish Nuclear Safety Council (CSN), Spain

Introduction

European Directive 90/641/Euratom established the general framework for operational protection of outside workers exposed to the risk of ionising radiation during their activities in controlled areas. In Spain this Directive has been transposed through Royal Decree 413/1997, which established specific responsibilities regarding Radiation Passbook and the training in radiation protection of outside workers.

The Spanish radiation passbook was introduced in 1990 and since this time, Nuclear Safety Council (CSN), as the regulatory authority has required that all outside workers entering controlled areas should have radiation passbooks. Recently, CSN has implemented improvements in the Spanish radiation passbooks taking into account the previous experience and the Directive 96/29/Euratom.

With regard to training in radiation protection of outside workers, in 1998 the CSN and the Association of the electricity utilities (UNESA) initiated joint efforts to define a new radiation protection training model for Spanish outside workers. This model should involve both the outside undertaking and the nuclear power plants. Several options already implemented in other European countries were studied and as result a model based on the following criteria was developed:

- An initial training course which is responsibility of the outside undertaking.
- A specific course which will be responsibility of the operators of the controlled area in which outside workers perform the activities.

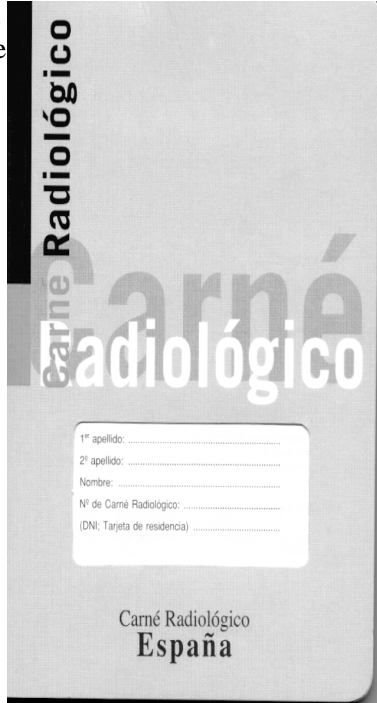
Radiation passbook

The operational protection of outside workers is regulated by the Royal Decree nº 413/1997, according this legislation all outside workers must have an individual radiological monitoring document (radiation passbook) containing all necessary information to ensure the radiological protection of these kind of workers (training information, health data, dose information).

The Royal Decree nº413/1997 empowered the Nuclear Safety Council to establish and to modify the format and content of the radiation passbook. So, the CSN performed an Instruction nºIS-01, which was published into national legislation on 31 May 2001.

This Instruction IS-01 defines the format and content of new Spanish radiation passbook. For designing the format of the new radiation passbook the CSN has taken into account the previous experience obtained since 1990 in using the past radiation passbook. So, the radiation passbook has been designed in such way that is possible to be filled in mechanically, ensures the reliability of the data based modifications of the data (coloured paper (beige) and sewing has been put in force since 1 of January of 2002.

data page



design of the new Spanish radiation passbook:

1. INFORMACIÓN A CUMPLIMENTAR POR LA EMPRESA EXTERNA EN LA APERTURA DEL CARNE

1.1. DOMICILIO DEL TITULAR (a cumplimentar también en caso de cambio de los datos iniciales):
 C/ Pea Volador n.º 3 Provincia: Madrid
 Población: Algete Teléfono: 91-773333 (proporvisivo (subrayar lo que proceda))
 C/
 Población: Provincia:
 Teléfono: (proporvisivo (subrayar lo que proceda))
 C/
 Población: Provincia:
 Teléfono: (proporvisivo (subrayar lo que proceda))

1.2. HISTORIAL DOSIMÉTRICO HASTA LA FECHA DE EMISIÓN DEL PRESENTE CARNE

1.2.1. DOSIS VIDA (mSv)

EXPOSICIÓN EXTERNA							CONTAMINACIÓN INTERNA	DOSIS EFECTIVA (suma de 1 y 2)	
(1) Dosis por persona	Dosis superiores	Dosis por exposición no uniforme					(2) Dosis efectiva compensada (Ei)		
		Manos	Arrozos	Pies	Tobillos	Costados	Otras zonas		
133	180	-	-	-	-	-	-	10	143

Observaciones (añadir aspectos relevantes en relación con las dosis recibidas por el trabajador como por ejemplo dosis resultantes de exposiciones especialmente autorizadas):

1.2.2. DOSIS EFECTIVA DE LOS ÚLTIMOS 5 (CINCO) AÑOS (sin contar año en curso)

AÑO	DOSIS EFECTIVA DE LOS ÚLTIMOS 5 (CINCO) AÑOS		OBSERVACIONES (1)
	EXPOSICIÓN EXTERNA (1)	CONTAMINACIÓN INTERNA (2)	
1997	35	35	
1998	8	8	
1999	25	25	
2000	10	10	
2001	7	7	

(1) Reflejar aspectos relevantes en relación con las dosis recibidas por el trabajador como por ejemplo dosis resultantes de exposiciones especialmente autorizadas.

El titular del presente carné, D. D. Juan Español Español declara la veracidad de la información referente a datos personales, condiciones y datos de exposición y sobre formación en protección radiológica anterior a la fecha de emisión del presente carné.

Firma del titular del carné

The new Spanish radiation passbook contains the following information:

- Outside worker's identity: includes the information necessary for the identification of the outside worker (name, surname, national identification document number or passport number, gender, date of birth, nationality and address).
- Dose information before the issue of the radiation passbook: effective dose equivalent from external exposure, from internal exposure and effective dose for the last five years.
- Outside undertakings (employer): name of the employer, national identification code, number in the CSN register for outside undertakings, dates of starting and finishing of the contract (for outside worker), occupational category, radiological classification, undertaking's stamp and signature of the undertaking responsible.
- Operators (facility): name of the facility, address, period of time covered by the activity, operator's stamp and signature of the operator responsible.
- Health data (medical surveillance): date and type of medical examination, medical classification, conditions for working and if any type of restriction for working applies, name and stamp of the approved occupational health service, national identification number and signature of the health physician.
- Basic and specific training: date, name of the training centre, training centre stamp and signature of centre's responsible person.

- Internal dosimetry: date of assignment (day/month/ year), date of measurement (month/year), type of control (routine, special), total activity (Bq), radionuclides, effective committed dose, approved dosimetry service's stamp, and signature of approved dosimetry service's responsible person.
- Dosimetric file which the operator (facility) must fill in: operational external dosimetry-time period, dose record, operator's stamp, and signature of the operators responsible.
- Dosimetric file which the undertaking (employer) must fill in:
 - Official external dosimetry-date (month and year), $H_p(10)$ and $H_p(0,07)$, internal dosimetry-committed doses, effective doses- official external ($H_p(10)$ plus internal doses), effective doses in the last five years, undertaking's stamp and signature of the undertaking responsible.
 - Doses of non-uniform exposure: date (month and year), dose equivalent in different parts of the body (hands, forearms, feet, ankles and others), undertaking's stamp and signature of the undertaking responsible.

Training in radiation protection for outside workers

The Spanish transposition of the European Directive through Royal Decree 413/97 distributed the responsibility for outside worker radiation protection training between the outside undertaking (basic training) and the nuclear and non-nuclear installations (as operator of controlled area is responsible to provide the outside worker specific training).

In 1998 was set up a joint *ad-hoc* Group, which included representatives from UNESA Radiological Protection and training divisions and from CSN for defining a radiation protection training model for outside workers. The members of this Working Group carried out a survey in European countries in order to acquire knowledge of the training models implemented in other countries in response to the European Directive. In February of 2001 the CSN and UNESA agreed a consolidated approach for radiation protection training of outside workers.

A series of general criteria were taking into account in the design of the training model which derived from the experience accumulated by training division at nuclear power plants:

- The training must be aimed at a wide sector of workers, the majority of whom are unqualified persons, so the contents should be simple and focusing on aspects relating to every day activity and of practical application.
- Specific training should be oriented basically towards practical aspects relating to knowledge of the nuclear plant, the radiological risks associated to the activities in which the worker would be involved and the methodology established for protection against these risks.

Taking into account these criteria two types of training have been established:

Basic training

- *Objectives*: to instruct workers on the specific characteristics of the risks involved in ionising radiation and on protection against such risks.

- *Responsibility*: Outside undertakings.
- *Programme*: The following topics have been considered for the basic training course:
 - Radiation and its effects.
 - Legal and administrative aspects.
 - Description of risks and protection measures: external exposure, external contamination, internal exposure, identification of risks and prevention measures.
 - Practical exercises and practical sessions.
- *Duration of the basic course*: Theoretical topics: 4,5 hours; practical topics: 1,5hours.
- *Evaluation*: An evaluation of the knowledge acquired has been considered necessary on completion of the course by means of a written test with 20 questions and a practical examination.
- *Accreditation*: Accreditation of successful completion of the course should be included in the outside worker's radiation passbook, filling in the training section.
- *Retraining*: Every two year the outside workers must attend a basic training course, and must undergo a written and practical examination, if the outside worker has reached the training objectives an accreditation should be included in the outside workers radiation passbook. This accreditation every two year included in the outside worker's radiation passbook is required as authorisation to work in a nuclear facility.

Specific training

- *Objectives*: to refresh and update the most important items learned during basic training and inform the worker of the specific radiological risks in the controlled areas, the measures to be taken and the specific performance standards.
- *Responsibility*: Operator of a controlled area in which outside workers perform activities.
- *Programme*: The following topics have been considered:
 - Physical concept and specific risk of the nuclear facility.
 - Plant risk prevention methodology.
 - Rules to be followed in controlled area.
 - Actions in event of radiological emergencies.
- *Duration of the specific course*: Theoretical topics: 4 hours.
- *Evaluation*: An evaluation of the knowledge acquired has been considered necessary on completion of the course by means of a written test with 20 questions.
- *Accreditation*: Accreditation of successful completion of the course should be included in the outside worker's radiation passbook, filling in the specific training section.

Current situation

After defining the design and contents on radiation protection training for outside workers, the CSN has drawn up a draft of an Instruction, which once will bring into force in national legislation, it will define the legal framework, content of the courses (basic and specific training), structure, needs of qualification and skills of the trainers and validity of the courses.

At the same time, UNESA is developing a pilot project for basic training on radiation protection for outside workers involve in activities in the nuclear facility field. The course is being designed around the learning outcomes required for working as outside workers in the nuclear field.

At first stage, a modular course will be developed with successive modules; Part 1 (Radiation and its effects), Part 2 (Legal and administrative aspects), Part 3 (Description of risks and protection measures). Practical activities in order to apply the theoretical knowledge to real situations have been incorporated into the appropriate modules. This course has been designed for face-to face teaching, including traditional methods as lectures, tutorials and structured practical sessions and contemporary methods such as videos and computer exercises.

Additionally to this course and at second stage, UNESA has planned to develop a course based on multi-media interactive programmes. This multi-media tool will consist of a relational database containing information about the same topics included in the modular course. The visualisation of this information will be done with a personal computer, and special attention will be given to the interactivity of the system which will support multiple choice questions. The system will validate the responses, distinguishes between the correct and the wrong answers.

References

- [1] European Commission Council Directive 90/641/EURATOM of 4 December 1990 on the Operational Protection of Outside Workers Exposed to the Risk of Ionizing Radiation during their activities in controlled areas. Official Journal of the European Communities L 349, 21 (13 December 1990).
- [2] “The Spanish National Dose Registry and Spanish Radiation Passbook” A. Hernandez, A. Martin, I. Villanueva, I. Amor, J. L. Butragueño. Radiation Protection Dosimetry Vol 96, Nos 1-3, pp 277-280 (2001).
- [3] “The new radiation protection training model for outside workers in Spain” I. Amor, I. Villanueva, J.A. Prieto, F. Massana and P. Carmena. Radioprotection – Colloques volume 35 C1(2000).
- [4] “Propuesta de programa de formación de trabajadores externos” I. Villanueva, I.Amor, J.A. Prieto, F. Massana, P. Carmena. VIII Congreso Nacional de la Sociedad Española de Protección Radiológica 27-29 Septiembre de 2000 (Maspalomas, Canarias).

LIST OF PARTICIPANTS

AUSTRIA

Mr. Abel GONZALEZ
IAEA
Division of Radiation and Waste Safety
Wagramerstrasse 5
PO BOX 100
A-1400 Vienna

Phone: +43 1 2600 22654/22526
Fax: +43 1 2600 29653
E-mail: a.gonzalez@iaea.org

Ms. Monica GUSTAFSSON
IAEA
Radiation Safety Section
PO Box 100
Wagramerstrasse 5
A-1400 Vienna

Phone: +43 1 2600 22725
Fax: +431 26007
E-mail: M.Gustafsson@iaea.org

Mr. Gabor LAFRANCO
Canberra Packard C.E. GmbH
Wienersiedlung 6
A-2432 Schwadorf

Phone: +43 2230 3700 35
Fax: +43 2230 3700 15
Email: glafranco@cpce.at

Mr. Seong Ho NA
IAEA
Radiation Safety Section
PO Box 100
Wagramerstrasse 5
A-1400 Vienna

Phone: +43 1 2600 22716
Fax: +431 26007
E-mail: S.Na@iaea.org

Mr. Alfred STRITZL
NuSAFE LLC
Radetzkystrasse 4/1
A-3100 ST. Polten

Tel/Fax: +43 2742 313 421
E-mail: ias@via.at

ARMENIA

Mr. Vovik ATOYAN
Head, Safety Department
Armenian NPP
Medzamor
Armavir Region, 377766

Phone: +3741 284232
Fax: +3741 151 860
E-mail: anpp@lx2.yerphi.am

BELGIUM

Mr. Luc VERMEULEN
Electrabel – Tihange NPP
Avenue de l'Industrie, 1
B-4500 Tihange

Phone: +32 85 24 33 51
Fax: +32 85 24 33 79
Email: luc.vermeulen@electrabel.com

BULGARIA

Mr. Georgi VALTCHEV
Kozloduy NPP
3321 Kozloduy

Phone: +359 973 72565
Fax: +359 973 80126
E-mail: valtchev@npp.cit.bg

CANADA

Mr. Arif KHAN
Ontario Power Generation
Darlington NGS
P.O. Box 160
1675 Montgomery Park Road
Pickering, Ontario L1V 2R5

Phone: +1 905 839 1151 ext. 3503
Fax: +1 905 837 7268
E-mail: arif.khan@ontariopowergeneration.com

CHINA

Mr. Jikai LI
Qinshan NPP
PO Box 602
Haiyan, Zhejiang 314300

Phone: +86 573 963 1989
Fax: +86 573 693 1368

Mr. Xinhe LIU
China National Nuclear Corporation
Radiation Protection
PO Box 2102 (14)
Beijing 100822

Phone: +86 10 6853 1494
Fax: +86 10 6853 9374
E-mail: liu401@yahoo.com

Mr. Junwu YANG
Lingao NPP
Daya Bay Shenzhen 518124
Haiyan, Zhejiang 314300

Phone: +86 755 442 9771
Fax: +86 755 447 9659

CZECH REPUBLIC

Mr. Josef KOC
Radiation Protection Dept. Manager
Temelin NPP
373 05 Temelin

Phone: +420 334 782 596
Fax: +420 334 782 794
E-mail: kocxxj1.ete@mail.cez.cz

Mr. Stanislav KOCVARA
VF, a.s.
Namesti Miru 50
679 21 Cerna Hora

Phone: +420 506 428 611
E-mail: office@vf.cz

Ms. Linda KVASNOVSKA
VF, a.s.
Business Dept.
Namesti Miru 50
679 21 Cerna Hora

Phone: +420 506 428 623
Fax: +420 506 428 630
E-mail: lin@vf.cz

Mr. Jiri MALYSAK
VF, a.s.
Namesti Miru 50
679 21 Cerna Hora

Phone: +420 506 428 611
E-mail: office@vf.cz

Ms. Karla PETROVA
State Office for Nuclear Safety
Senovážné náměstí 9
110 00 Prague 1

Phone: +420 2 2162 4556
Fax: +420 2 2162 4710
Email: karla.petrova@sujb.cz

FINLAND

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
E-mail: timo.kontio@fortum.com

Mr. Erik LEHTONEN
RADOS Technology Oy
PO Box 506
20101 Turku

Phone: +358 2 468 4660
Fax: +358 2 468 4601

Mr. Veli RIIHILUOMA
Finnish Centre for Radiation
and Nuclear Safety Authority, STUK
PO Box 14
00881 Helsinki

Phone: +358 9 7598 8310
Fax: +358 9 7598 8382
Email: veli.riihiluoma@stuk.fi

Ms. Suvi RISTONMAA
STUK
Radiation and Nuclear Safety Authority
PO Box 14
00881 HELSINKI

Phone: +358 9 759 88442
Fax: +358 9 759 88382
E-mail: suvi.ristonmaa@stuk.fi

FRANCE

Mr. Marc AMMERICH
DGSNR - SD4
10, Route du Panorama
BP 83
F-92266 Fontenay-aux-Roses Cedex

Phone: +33 143 19 70 40
Fax: +33 143 19 71 30
E-mail: marc.ammerich@asn.minefi.gouv.fr

Ms. Dominique ARNAUD ASN-DRIRE-PACA 67-69, Avenue du Prado F-13286 Marseille Cedex 6	Phone: +33 4 91 83 63 22 Fax: +33 4 91 83 44 10 E-mail: dominique.arnaud@asn.minefi.gouv.fr
Ms. Catherine BAILLOEUIL EDF/GDF – SCAST/Pôle MAN 22-28, rue Joubert F-75009 Paris	Phone: +33 1 55 31 47 13 Fax: +33 1 55 31 47 99 E-mail: catherine.bailloeuil@edf.fr
Ms. Corinne BENHAMOU CEPN BP 48 – Route du Panorama F-92263 Fontenay-aux-Roses Cedex	Phone: +33 146 54 79 09 Fax: +33 140 84 90 34 E-mail: benhamou@cepn.asso.fr
Mr. Marc CHAMPION IRSN/DPHD/SEGR/SAER BP 17 F-92262 Fontenay-aux-Roses Cedex	Phone: +33 1 46 54 71 40 Fax: +33 1 46 54 88 29 E-mail: marc.champion@irsn.fr
Mr. Philippe COLSON EDF-DPN Groupe Prévention des Risques Cap Ampère – Bâtiment Tête Pleyel 1, Place Pleyel F-93282 Saint-Denis Cedex	Phone: +33 143 69 31 07 Fax: +33 143 69 45 70 E-mail: philippe.colson@edf.fr
Ms. Lucie D'ASCENZO CEPN BP 48 – Route du Panorama F-92263 Fontenay-aux-Roses Cedex	Phone: +33 146 54 76 60 Fax: +33 140 84 90 34 E-mail: dascenzo@cepn.asso.fr
Mr. Jean-Pierre DEGRANGE CEPN BP 48 – Route du Panorama F-92263 Fontenay-aux-Roses Cedex	Phone: +33 146 54 81 41 Fax: +33 140 84 90 34 E-mail: degrange@cepn.asso.fr
Mr. Jean-Michel DELIGNE IRSN/DPHD/SEGR/SAER BP 17 F-92262 Fontenay-aux-Roses Cedex	Phone: +33 1 46 54 86 23 Fax: +33 1 46 54 88 29 E-mail: jean-michel.deligne@irsn.fr
Mr. Jean-Louis FENOLLAND Médecin du Travail 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
Responsable commercial Europe
MGP Instruments
Route d'Eyguières
BP 1
F-13113 Lamanon

Phone: +33 4 90 59 60 34
Fax: +33 4 90 59 55 18
Email: rforest@mgpi.com.fr

Mr. Yves GARCIER
EDF/Production nucléaire
Pôle Industrie
Site Cap Ampère – Tête Pleyel B
1 place Pleyel
F-93282 Saint-Denis Cedex

Phone: +33 143 69 30 20
Fax: +33 143 69 34 95
E-mail: yves.garcier@edf.fr

Ms. Michèle GONIN
EDF/GDF – SCAST/Pôle MAN
22-28, rue Joubert
F-75009 Paris

Phone: +33 155 31 47 10
Fax: +33 155 31 47 99
E-mail: michele.gonin@edf.fr

Mr. Jean-François LABOUGLIE
EDF/CNPE de Cattenom
Chargé de Mission Radioprotection
BP 41
F-57570 Cattenom

Phone: +33 3 82 51 78 01
Fax: +33 3 82 54 40 30
E-mail: jean-francois.labouglie@edf.fr

Mr. Jacques LEBEAU
EDF/DPN - Projet ALARA
1 Place Pleyel – Cap Ampère
F-93282 Saint-Denis Cedex

Fax: +33 143 69 45 70
E-mail: jacques.lebeau@edf.fr

Mr. Christian LEFAURE
CEPN
BP 48 – Route du Panorama
F-92263 Fontenay-aux-Roses Cedex

Phone: +33 146 54 79 08
Fax: +33 140 84 90 34
E-mail: lefaure@cepn.asso.fr

Mr. Stefan MUNDIGL
OECD/NEA
Radiation Protection and
Radioactive Waste Management Division
Le Seine St. Germain 12, bld des Iles
F-92130 Issy-les-Moulineaux

Phone: +33 145 24 10 45
Fax: +33 145 24 11 10
E-mail: mundigl@nea.fr

Mr. Pierre PIRO
EDF/CNPE du Blayais
BP 27
F-33820 St.-Ciers-sur-Gironde

Phone: +33 5 57 33 28 27
Fax: +33 5 57 33 32 49
E-mail: pierre.piro@edf.fr

Mr. Alain ROCHER
EDF – Groupe Prévention des Risques
CAP AMPERE – Bâtiment Tête Pleyel
1, Place Pleyel
F-93282 Saint-Denis Cedex

Phone: +33 143 69 31 56
Fax: +33 143 69 45 71
E-mail: alain.rocher@edf.fr

Mr. Alain ROUPIOZ
EDF/DPN – Groupe Prévention des Risques
Cap Ampère
1, Place Pleyel
F-93282 Saint Denis Cedex

Phone: +33 143 69 31 42
Fax: +33 143 69 45 71
E-mail: alain.roupioz@edf.fr

Mr. David TAABANI
EDF – CNPE Gravelines
BP 149
F-59820 Gravelines

Phone: 33 3 28 68 40 00
Fax: +33 3 28 68 41 19
E-mail: david.taabani@edf.fr

Mr. Christian THEZEE
EDF/Présidence/IGSN
21, avenue de Messine
F-75008 Paris

Phone: 33 1 40 42 36 95
Fax: +33 1 40 42 75 89
E-mail: christian.thezee@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
E-mail: dominique-rene.vincent@edf.fr

Mr. Jean-Marc WILLEMENOT
EDF/DPN - GENV Bureau D03/34
Site Cap Ampère
1, Place Pleyel
F-93282 Saint-Denis Cedex

Phone: +33 1 43 69 31 43
Fax: +33 1 43 69 45 71
E-mail: jean-marc.willemenot@edf.fr

GERMANY

Mr. Ingolf BRIESEN
Kernkraftwerk Obrigheim GmbH
Postfach 1161
D-74847 Obrigheim

Phone: +49 6261 65 380 (stand)
+49 6261 65361
Fax: +49 6261 65490
E-mail: kwo.i.briesen@kwobrigheim.de

Mr. Stephan HÜBNER
GSF – Research Center
Ingolstädter Landstr. 1
D-85764 Neuherberg

Phone: +49 89 3187 2350
Fax: +49 89 3187 3328
E-mail: stephan.huebner@gsf.de

Mr. Heinz Peter KAPTEINAT
VGB – Geschäftsstelle
Postfach 10 39 32
D-45039 Essen

Phone: +49 201 81 28 248
Fax: +49 201 81 28 345
E-mail: heinzpeter.kapteinat@vgb.org

Mr. Manfred MEYER
Kernkraftwerk Philippsburg GmbH (KKP)
Rheinschanzinsel
Postfach 1140
D-76652 Philippsburg

Phone: +49 7256 95 2250
Fax: +49 7256 95 3488
E-mail: m.meyer@kkp.enbw.com

Mr. Dieter PASTOR
Framatome ANP GmbH
Seligenstädter Str. 100
D-63791 Karlstein

Phone: +49 6188 780 9726
Fax: +49 6188 780 9666
E-mail: dieter.pastor@framatome-anp.de

Mr. Wolfgang PFEFFER
Gesellschaft für Anlagen-
und Reaktorsicherheit (GRS) mbH
Schwertnergasse 1
D-50667 Köln

Phone: +49 221 2068 773
Fax: +49 221 2068 888
E-mail: pff@grs.de

Mr. Frank SCHATMANN
BRENK Systemplanung GmbH
Heider-Hof-Weg 23
D-52080 Aachen

Phone: 49 2405 4651 26
Fax: +49 2405 4651 50
E-mail: f.schartmann@brenk.com

Mr. Josef SCHOBER
TÜV Süddeutschland Bau und Betrieb GmbH
BB-EKS 3
Westendstrasse 199
D-80686 MÜNCHEN

Phone: +49 89 5791 1562
Fax: +49 89 5791 1594
E-mail: josef.schober@tuev-sued.de

Mr. Ian Roger TERRY
Framatome-ANP GmbH
Dept. NDS4 – Building 34
Freyeslebenstr. 1
D-91058 Erlangen

Phone: +49 9131 18 96336
Fax: +49 9131 18 96835
E-mail: Ian.Terry@framatome-anp.de

Mr. Günter WEIMER
RSK-Geschaefsstelle
c/o Bundesamt für Strahlenschutz
Postfach 12 06 29
D-53048 Bonn

Phone: +49 228 305 3720
Fax: +49 228 670 388
E-mail: gweimer@bfs.de

Mr. Kjell WESTERBERG
Sina Industrieservice GmbH & Co. KG
Bayernstrasse 53
D-75177 Pforzheim

Phone: +49 7231 380801
Fax: +49 7231 380855
E-mail: kjellwesterberg@sina.de

HUNGARY

Mr. Sandor DEME
KFKI Atomic Energy Research Institute
PO Box 49
1525 Budapest

Phone: +36 1 392 2291
Fax: +36 1 395 9293
E-mail: deme@sunserv.kfki.hu

Mr. Barnabas NAGY
Hungarian Atomic Energy Authority
Margit krt. 85
PO Box 676
1539 Budapest

Phone: +36 75 507 898
Fax: +36 75 311 471
E-mail: nagy@haea.gov.hu

Mr. Gabor VOLENT
PAKS Nuclear Power Plant Ltd.
Radiation Protection Department
P.O. Box 71
7031 Paks

Phone: +36 75 507476
Fax: +36 75 507390
E-mail: volent@npp.hu

KOREA

Mr. Wan Joo KIM
Korea Institute of Nuclear Safety
19 Kusong, Yusong
305-338 Taejon

Phone: +82 42 868 0583
Fax: +82 42 868 0531
E-mail: k391kwj@kins.re.kr

Mr. Sae-Tae KIM
Ulchin NPP
Units 1 & 2
84-4, Bugu-Ri, Buk-Myun, Ulchin-Gun
Kyung-Buk 767890

Phone: 82 565 785 1370
Fax: +82 565 785 1599
E-mail: kimseta@khnp.co.kr

Mr. Byoung-Chan LIM
Safety & Technology Department
Korea Hydro & Nuclear Power Company
PO Box 167
Samsung-Dong, Kangnam-Gu
Seoul

Phone: +82 2 3456 2434
Fax: +82 2 3456 2469
E-mail: imbycha@khnp.co.kr

LITHUANIA

Mr. Gintautas KLEVINSKAS
Radiation Protection Centre
Kalvariju 153
2042 Vilnius

Phone: +370 2 778651 or 754 691
Fax: +370 2 754692
E-mail: g.klevinskas@rsc.lt

Mr. Viktor PLETNIOV
Ignalina NPP
4761 Visaginas

Phone: +370 662 8828
Fax: +370 662 9350
E-mail: pletnev@mail.iae.lt

LUXEMBOURG

Dr. Jochen NAEGELE
Commission Européenne
DG Environnement C4
Bât. Jean Monnet, WAG C3/347
2920 Luxembourg

Phone: +352 4301 36027
Fax: +352 4301 34646
E-mail: jochen.naegele@cec.eu.int

Mr. Klaus SCHNUER
Commission Européenne
DG Environnement A-1
Centre A. Wagner C331, Rue A. de Gasperi
Plateau de Kirchberg
2920 Luxembourg

Phone: +352 4301 3 6388
Fax: +352 4301 3 4646
E-mail: 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
E-mail: F.kamping@epz.nl

Mr. Frits VAN DE VELDE
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

Mr. Bob VAN DER WERF
Ministry for Environment
VROM Inspectie
Kernfysische Dients/IPC 560
Postbus 16191
NL-2500 BD DEN HAAG

Phone: +31 70 3392568
Fax: +31 70 339 1887
E-mail: Bob.vanderWerf@minvrom.nl

PAKISTAN

Mr. Jamal AFZAL
Chashma NPP
Maintenance Department
PO Box Chashma 42030
Chashma Barrage Colony
Mianwali

Phone: +92 45 202 41687
Fax: +92 45 202 41505
E-mail: coppcl@paknet2.ptc.pk

Ms. Bushra NASIM
Directorate of Nuclear Safety
and Radiation Protection
Pakistan Atomic Energy Commission (PAEC)
PSO
P.O. Box 1912
Islamabad

Phone: +92 51 920 2618/313
Fax: +92 51 920 4112
E-mail: bushra_nasim@hotmail.com

Mr. Ali ZULFIQAR
Safety & Health Physics Division
Chashma NPP
Maintenance Department
PO Box Chashma 42030
Chashma Barrage Colony
Mianwali

Phone: +92 45 202 41730-35 ext. 4072
Fax: +92 45 202 41505
E-mail: coppc1@paknet2.ptc.pk

RUSSIAN FEDERATION

Mr. Boris BEZRUKOV
Rosenergoatom
Ministry of the Russian Federation
for Atomic Energy
Bolshaya Ordynka 24/26
109017 Moscow

Phone: +7 095 220 62 45
Fax: +7 095 220 62 45
E-mail: bezrukov@rea.x-atom.ru

Mr. Alexandre EPIKHINE
Leningrad NPP
Sosnovy Bor
Leningrad Region 188540

Phone: +7 812 692 3215
Fax: +7 812 692 2518
E-mail: zgib@laes.sbor.ru

Mr. Alexander PETROV
Balakovo NPP
413840 St. Stepnaya
Balakovo – Saratov Region

Phone: +7 845 70 30237
Fax: +7 845 70 32638
E-mail: npp@aes.encred.ru

ROMANIA

Mr. Alexandru Leonte RODNA
National Commission for
Nuclear Activities Control
14 Libertatii Blvd.
42-4 Bucharest-5

Phone: +40 1 410 27 54 ext. 207
Fax: +40 1 411 14 36
E-mail: alexandru.rodna@cncan.ro

Mr. Vasile SIMIONOV
Societatea Nationala “Nuclearelectrica” SA
1 Medgidiei Street
Cernavoda 8625

Phone: 40 41 239 646
Fax: +40 41 239 269
E-mail: vsimionov@cne.ro

SLOVAKIA

Mr. Lubomir DOBIS
Bohunice NPP
Radiation Protection Department
Atomové Elektrárne
919 31 Jaslovske Bohunice

Phone: +421 33 597 2128
Fax: +421 33 597 4786
E-mail: dobis_lubomir@ebo.seas.sk

Mr. Jan KAIZER
Bohunice NPP
919 31 Jaslovske Bohunice

Phone: +42 1 33 597 3673
Fax: +42 1 33 597 4786
E-mail: kaizer_jan@ebo.seas.sk

Mr. Jaroslav SVITEK
Bohunice NPP
Atomové Elektrárne
919 31 Jaslovské Bohunice

Phone: +421 33 597 3298
Fax: +421 33 597 4786
E-mail: 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
E-mail: viktory@szusr.sk

SLOVENIA

Mr. Borut BREZNIK
Krsko NPP
Radiation Protection Department
Vrbina 12
8270 Krsko

Phone: +386 7 4802 287
Fax: +386 7 49 21 006
E-mail: borut.breznik@nek.si

Ms. Milena CERNILOGAR-RADEZ
Slovenian Nuclear safety Administration
Ministry of the Environment and Spatial Planning
Vojkova 59
1113 Ljubljana

Phone: +386 1472 1100
Fax: +386 1472 1199
E-mail: milena.cernilogar-radez@gov.si

Ms. Helena JANZEKOVIC
Slovenian Nuclear safety Administration
Ministry of the Environment and Spatial Planning
Vojkova 59
1113 Ljubljana

Phone: +386 1472 1194
Fax: +386 1472 1199
E-mail: helena.janzekovic@gov.si

Ms. Nina JUG
Health Inspectorate
of Republic of Slovenia
Parmova 33
1000 Ljubljana

Phone: 386 12803821
Fax: +386 12803808
E-mail: nina.jug@gov.si

Mr. Matjaz KOZELJ
"Jozef Stefan" Institute
Nuclear Training Centre
Jamova 39
PO Box 3000
1001 Ljubljana

Phone: +386 15885 277
Fax: +386 15612 276
E-mail: matjaz.kozelj@ijs.si

Mr. Milko Janez KRIZMAN
Slovenian Nuclear safety Administration
Ministry of the Environment and Spatial Planning
Vojkova 59
1113 Ljubljana

Phone: +386 1472 1155
Fax: +386 1472 1199
E-mail: milko.krizman@gov.si

Mr. Gregor OMAHEN
ZVD – Institute of Occupational Safety
Bohoriceva ulica 22a
SI-1000 Ljubljana

Phone: +386 1 43 20 253
Fax: +386 1 312 562
E-mail: gregor.omahen@zvd.si

Mr. Marjan PAVLIN
Krsko NPP
Vrbina 12
8270 Krsko

Phone: +386 7 4802 691
Fax: 386 7 4821 528
E-mail: marjan.pavlin@nek.si

Mr. Joze REBERSEK
Krsko NPP
Vrbina 12
8270 Krsko

E-mail: joze.rebersek@nek.si

Mr. Stane ROZMAN
Director of Krsko NPP
Krsko NPP
Vrbina 12
8270 Krsko

Ms. Dunja SEVER
Chief Health Inspector
of the Republic of Slovenia
Parmova 33
1000 Ljubljana

Phone: +386 1 280 3802
Fax: +386 1 280 3808
E-mail: dunja.sever@gov.si

Mr. Damijan SKRK
Health Inspectorate
of Republic of Slovenia
Parmova 33
1000 Ljubljana

Phone: + 386 1280 3823
Fax: +386 1280 3808
E-mail: Damijan.skrk@gov.si

Mr. Bojan SUSIN
Krsko NPP
Vrbina 12
8270 Krsko
Email: bosan.susin@nek.si

E-mail: bosan.susin@nek.si

Mr. Tomaz SUTEJ
Health Inspectorate of Republic of Slovenia
Parmova 33
1000 Ljubljana

Phone: +386 1280 3818
Fax: +386 1280 3808
E-mail: tomaz.sutej@gov.si

Mr. Ales VOLCANSEK
Krsko NPP
Vrbina 12
8270 Krsko

Phone: +386 7 4802 705
Fax: +386 7 49 21 528
E-mail: ales.volcansek@nek.si

Mr. Urban ZDESAR
ZVD – Institute of Occupational Safety
Bohoriceva ulica 22a
SI-1000 Ljubljana

Phone: +386 1 43 20 253
Fax: +386 1 312 562
E-mail: urban.zdesar@zvd.si

Mr. Marjan ZORKO
KRSKO NPP
Radiation Protection Department
Vrbina 12
8270 Krsko

Phone: +386 74 4802 000
E-mail: marjan.zorko@nek.si

SPAIN

Mr. Ramiro FRAGIO RODRIGUEZ
Central Nuclear de Cofrentes
46625 Cofrentes (Valencia)

Phone: +34 96 189 4300
Fax: +34 96 219 6477
E-mail: rjfragio@iberdola.es

Mr. Francisco José GONZALEZ TARDIU
Central Nuclear de Asco
Radiological Protection Service
43791 Asco (Tarragona)

Phone: +34 977 41 50 00
Fax: +34 977 405181
E-mail: fgtardiu@anacnv.com

Ms. Olvido GUZMAN
Consejo de Seguridad Nuclear
Justo Dorado, 11
28040 Madrid

Phone: +34 91 3460 653
Fax: +34 91 3460 588
E-mail: ogl@csn.es

Ms. Maria-Jesus MUNOZ
Consejo de Seguridad Nuclear
Justo Dorado, 11
28040 Madrid

Phone: +34 91 3460 410
Fax: +34 91 3460 588
E-mail: mjmglez@csn.es

Mr. Manuel RODRIGUEZ MARTI
Consejo de Seguridad Nuclear (CSN)
General Deputy of Operational Radiation Protection
C/Justo Dorado N°11
28040 Madrid

Phone: +34 91 3460136 or 60171
Fax: +34 91 3460588
E-mail: mrm@csn.es

Ms. Maria Luisa ROSALES
Consejo de Seguridad Nuclear
Justo Dorado, 11
28040 Madrid

Phone: +34 91 3460 128
Fax: +34 91 3460 588
E-mail: mlrc@csn.es

Ms. Isabel VILLANUEVA DELGADO
Consejo de Seguridad Nuclear
Operational Radiological Protection Deputy
Justo Dorado, 11
28040 Madrid

Phone: +34 91 3460 637
Fax: +34 91 3460 588
E-mail: ivd@csn.es

SWEDEN

Mr. Mikael ANDERSSON
Westinghouse Atom AB
Nuclear Fuel Operations
Regulatory affairs and safety
S-721 63 Vasteras

Phone: +46 21 347 714
Fax: +46 21 348005
E-mail: mikael.r.andersson@se.westinghouse.com

Mr. Bengt BENGTTSSON
Health Physics Supervisor
Ringhals Nuclear Power Plant
Unit 3 & 4
S-430 22 Väröbacka

Phone: +46 340 66 7967
Fax: +46 340 667 305
E-mail: bengt.bengtsson@ringhals.se

Ms. Eva FREDRIKSSON
Westinghouse Atom AB
Fuel Services – BWR Services
S-721 63 Västeras

Phone: +46 21 347 229
Fax: +46 21 347 398
E-mail: eva.fredriksson@se.westinghouse.com

Mr. Staffan HENNIGOR
Forsmark NPP
Forsmarks Kraftgrupp AB
S-742 03 Östhammar

Phone: +46 17382076
Fax: +46 173 81850
E-mail: sig@forsmark.vattenfall.se

Mr. Kjell LINDBERG
Forsmark NPP
Forsmarks Kraftgrupp AB
S-742 03 Östhammar

Phone: +46 173 81505
Fax: +46 173 82417
E-mail: krl@forsmark.vattenfall.se

Mr. Carl Göran LINDVALL
Manager Health Physics
Barsebäck Kraft AB
Barsebäck Nuclear Power Plant
Box 524
S-246 25 Löddeköpinge

Phone: +46 4672 4350
Fax: +46 46 724 580
E-mail: 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
E-mail: ingemar.lund@ssi.se

Mr. Christer SOLSTRAND
OKG Aktiebolag
S-572 83 Oskarshamn

Phone: +46 491 786 406
Fax: +46 491 78 78 15
E-mail: christer.solstrand@okg.sydskraft.se

Mr. Torgny SVEDBERG
Ringhals AB
Ringhals 1 NPP
Dept. R1ML
S-430 22 Väröbacka

Phone: +46 340 66 72 60
Fax: +46 340 66 51 02
E-mail: torgny.svedberg@ringhals.se

Mr. Jack VALENTIN
Scientific Secretary
International Commission on
Radiological Protection (ICRP)
S-171 16 Stockholm

Phone: +46 8 729 7275
Fax: +46 8 729 7298
E-mail: jack.valentin@ssi.se

SWITZERLAND

Mr. Swen Gunnar JAHN
Swiss Federal Nuclear Safety
Inspectorate HSK
Sektion Radiologischer Arbeitsschutz
Koordination Nuklearer Forschungsanlagen
CH-5232 Villigen - HSK

Phone: +41 56 310 3931
Fax: +41 56 310 4941
E-mail: jahn@hsk.psi.ch

Mr. Harald WELLER
Kernkraftwerk Gosgen-Daniken AG
Postfach 55
CH-4658 DANIKEN

Phone: +41 62 288 20 00
Fax: +41 62 288 20 01
E-mail: hweller@kkg.ch

UKRAINE

Ms. Tetyana LISOVA
Ministry of Fuel and Energy of Ukraine
State Nuclear Energy Department
Radiation Safety Division
30 Khreshchatyk Str.
01601 Kyiv-1

Phone: +380 44 229 2742
Fax: +380 44 462 0561
E-mail: lisova@mintop.energy.gov.ua

Mr. Volodymir NOVIKOV
National Nuclear Energy Generating Co.
“Energoatom”
9/11 Arsenalna str.
01011 KYIV-11

Phone: +380 44 294 4875
Fax: +380 44 294 4882

UNITED KINGDOM

Ms. Wendy BINES
HSE
Health Directorate B
Ionising Radiation Policy Unit
6 NW Rose Court
2 Southwark Bridge
London SE1 9HS

Phone: +44 20 7717 6831
Fax: +44 20 7717 6681/6199
E-mail: wendy.bines@hse.gsi.gov.uk

Mr. Michael BONE
British Energy Generation Ltd.
Operations Division
Sizewell B Power Station
NR Leiston
Suffolk, IP16 4UR

Phone: +44 1728 653 181
Fax: +44 1728 653 320
E-mail: michael.bone@british-energy.com

Mr. Robert FLETCHER
Siemens Environmental Systems Limited
Sopers Lane, Poole
Dorset BH17 7ER

Phone: +44 1202 782780
Fax: +44 1202 782056
E-mail: robert.fletcher@poole.siemens.co.uk

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 653189
E-mail: matthew.lunn@british-energy.com

Mr. Simon MORRIS
British Energy Generation Ltd.
Health and Safety Department (HSED)
Barnett Way
Barnwood
Gloucester, GL4 3RS

Phone: +44 1452 652 867
Fax: +44 1452 652 509
E-mail: simon.morris@british-energy.com

Mr. Andy R. WEEKS
BNFL – MAGNOX
Research and Technology (R & T)
Berkeley Centre
Gloucestershire GL13 9PB

Phone: +44 1453 81 28 55
Fax: +44 1453 81 21 12
E-mail: andy.r.weeks@magnox.co.uk

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
E-mail: tasos.zodiates@british-energy.com

UNITED STATES

Mr. Richard DOTY
PPL Susquehanna, LLC
Supervisor, Operations Technology
2, North Ninth Street (GENA 93)
Allentown, PA 18101

Phone: +1 610 774 7932
Fax: +1 610 774 7205 or 7337
E-mail: rldoty@pplweb.com

Ms. Harriet KARAGIANNIS
U.S. Nuclear Regulatory Commission
Radiation Protection Environmental Risk
and Waste Management Branch
Mail Stop T-9 F31
Washington, DC 20555

Phone: +1 301 415 6377
Fax: +1 301 415 5389
E-mail: hxx@nrc.gov

Mr. David MILLER
ISOE North American Technical Center
College of Engineering MC-234
206 Nuclear Engineering Lab
University of Illinois
103 S. Goodwin Ave.
Urbana, Illinois 61801-2984

Phone: +1 217 333 1098
Fax: +1 217 333 2906
E-mail: dwmphd@aol.com

Mr. Kenneth OHR
Quad Cities NPP
Radiation Eng. Manager
Exelon Corporation
22710 206th Avenue North
Cordova, IL 61242-9740

Phone: +1 309 227 2725
Fax: +1 309 227 2265
E-mail: kenneth.ohr@exeloncorp.com

Mr. R. Scott SCHOFIELD
San Onofre Nuclear Generating Station
Southern Californian Edison Co.
PO BOX 128
Mail Stop D1N
San Clemente, CA 92674-0128

Phone: +1 949 368 6164
Fax: +1 949 368 6049
E-mail: schofirs@songs.sce.com

Mr. David WOOD
DC Cook NPP, American Electric Power
One Cook Place
Bridgeman, MI. 49106

Phone: +1 616 466 2559
Fax: +1 616 465 6061
E-mail: wdwood@aep.com

OECD PUBLICATIONS, 2 rue André-Pascal, 75775 PARIS CEDEX 16
Printed in France.