IAEA-TECDOC-1572

# Disposal Aspects of Low and Intermediate Level Decommissioning Waste

Results of a coordinated research project 2002–2006

# **CD-ROM**



December 2007

The originating Section of this publication in the IAEA was:

Waste Technology Section International Atomic Energy Agency Wagramer Strasse 5 P.O. Box 100 A-1400 Vienna, Austria

DISPOSAL ASPECTS OF LOW AND INTERMEDIATE LEVEL DECOMMISSIONING WASTE IAEA, VIENNA, 2007 IAEA-TECDOC-1572 ISBN 978–92–0–109107–9 ISSN 1011–4289

© IAEA, 2007

Printed by the IAEA in Austria December 2007

# EDITORIAL NOTE

The papers in these proceedings are reproduced as submitted by the authors and have not undergone rigorous editorial review by the IAEA.

The views expressed do not necessarily reflect those of the IAEA, the governments of the nominating Member States or the nominating organizations.

The use of particular designations of countries or territories does not imply any judgement by the publisher, the IAEA, as to the legal status of such countries or territories, of their authorities and institutions or of the delimitation of their boundaries.

The mention of names of specific companies or products (whether or not indicated as registered) does not imply any intention to infringe proprietary rights, nor should it be construed as an endorsement or recommendation on the part of the IAEA.

The authors are responsible for having obtained the necessary permission for the IAEA to reproduce, translate or use material from sources already protected by copyrights.

#### **CONTENTS OF THE CD-ROM**

#### ARGENTINA

Disposal aspects of Ra-1 research reactor decommissioning waste

#### CANADA

Decommissioning of Ontario power generation's nuclear stations — radioactive waste volume arisings and radionuclide inventories

### CHINA

Feasibility study of the option of on-site disposal of decommissioned nuclear wastes in China

#### GERMANY

Management of decommissioning waste in Germany

#### HUNGARY

Investigations of the disposal aspects for low and intermediate level wastes originating from the decommissioning of Paks NPP

#### INDIA

Studies for onsite disposal of waste from decommissioning/revamping of nuclear facilities and NPPs in India

#### REPUBLIC OF KOREA

A study on the disposal concept and packaging of low- and intermediate-level decommissioning waste in Korea

#### LITHUANIA

Disposal aspects of low and intermediate level Ignalina NPP decommissioning waste

#### **RUSSIAN FEDERATION**

Research project of Saint Petersburg institute of technology: the main results

#### SLOVAKIA

Impact of a1 decommissioning waste disposal on repository performance assessment methodology development

#### **SWEDEN**

Performance assessment of a planned near-surface repository for short lived waste from decommissioning of Swedish nuclear power plants

#### UNITED KINGDOM

Aspects of packaging metallic radioactive waste for disposal

#### UKRAINE

Decommissioning waste management strategy in ukraine

# UNITED STATES OF AMERICA

Challenges in source term modelling of decontamination and decommissioning wastes

CONTRIBUTORS TO DRAFTING AND REVIEW

#### ARGENTINA

#### DISPOSAL ASPECTS OF RA-1 RESEARCH REACTOR DECOMMISSIONING WASTE

# S. Harriague, C. Barberis, E. Cinat, C. Grizutti, H. Scolari,

Comisión Nacional de Energía Atómica, Buenos Aires

# Abstract

The objective of the project is to analyze disposal aspects of waste from total dismantling of Argentinean research reactors, starting with the oldest one, 48 years old RA-1.

In order to estimate decommissioning waste, data was collected from files, area monitoring, measurements, sampling to measure activity and composition, operational history and tracing of operational incidents. Measurements were complemented with neutron activation calculations.

Decommissioning waste for RA-1 is estimated to be 71.5 metric tons, most of it concrete (57 tons), the rest being steels, lead and reflector graphite (4.8 tons).

Due to their low specific activities, no disposal problems are foreseen in the case of metals and concrete. Disposal of aluminium, steel, lead and concrete is analyzed. On the contrary, as the country has no experience in managing graphite radioactive waste, work was concentrated on that material.

Stored (Wigner) energy may exist in RA-1 graphite reflectors irradiated at room temperature. Evaluation of stored energy by calorimetric methods is proposed, and its annealing by inductive heating; HEPA filters should be used to deal with gaseous activity emissions, mainly Cl-36 and C-14.

Galvanic corrosion, dust explosion, ignition and oxidation can be addressed and should not become disposal problems. Care must be taken with graphite dust generation and disposal, due to wetting and flotation problems.

Lessons learned from the project are presented, and the benefits of sharing international experience are stressed.

### 1. SCOPE

# 1.1. Background

Large nuclear facilities in Argentina include two operating NPPs and a third one under construction, three research reactors, a radioisotope production reactor, two critical assemblies, irradiation plants, several facilities for radioisotope source production and industrial/medical applications, a Mo-99 production plant, large accelerators and several fuel cycle facilities, including power reactor and research reactor fuel production plants.

Low level operational waste from these facilities has been disposed in the past in solid and liquid trenches. During the last 50 years, experience has been gained on management of different types of radioactive waste. At present, the site selection and conceptual design of low level and intermediate level waste repositories are under way.

The National Programme on Radioactive Waste Management, run by the National Atomic Energy Commission –CNEA, requires estimates of decommissioning waste and analysis of its disposal aspects as input for the design of the low and intermediate level waste repositories and for establishing their acceptance requirements. This study begins with RA-1, a 40 kw pool reactor that reached criticality in January 1958 being the oldest relevant nuclear facility in the country.

# **1.2.** Objective of the project

The main purpose of the Project is to analyze, based on the RA-1 case, disposal aspects of decommissioning waste from the Argentinean research reactors, and to propose disposal alternatives whenever necessary.

The amount and type of decommissioning waste arising from total dismantling of the RA-1 reactor will be estimated. The procedure for waste estimation and the analysis of disposal alternatives will be applied to the other research reactors in the country.

Participation in an IAEA Coordinated Research Project will allow the reader to learn from international experience and consequently to develop human resources.

# 1.3. Methodology

Information and data on reactor design, modifications, materials, irradiation history and operational incidents were obtained from documents and from operation staff; retired personnel having participated in the design, construction and operation of the facility was also contacted. [1]

The procedure used to estimate the decommissioning waste was as follows:

- Collect data from area monitoring.
- Measurements on accessible areas (reactor is operational).
- Sampling and  $\alpha$ ,  $\beta$  and  $\gamma$  activity measurements.
- Analysis of operational incidents that may have lead to contamination.
- Neutron activation calculation, in turn requiring knowledge of materials composition.

Materials composition was obtained from records and measurements on accessible materials. Residual elements in metals were obtained from records and from analysis of the country production practices at the time of the reactor construction. Composition and impurity concentrations in concrete, graphite and lead, where little information has been kept, were obtained from chemical analysis, neutron activation analysis and estimations based on a wide literature survey.

The low and intermediate level waste arising from total dismantling of the RA-1 reactor is estimated from the calculations and measurements. Based on the country experience, waste presenting unusual disposal issues is identified, and disposal alternatives are developed.

### 2. RESULTS

Main results from the project are the following.

### 2.1. Decommissioning waste

Table 1 shows primary decommissioning waste from total dismantling of the reactor; both activated and contaminated materials are included. For the evaluation of conditioned volumes, maximum package densities of 1250 kg/m<sup>3</sup> for metals, and 1100 kg/m<sup>3</sup> for both concrete rubble and graphite, were assumed:

Material	Mass (kg)	Conditioned Volume (m <sup>3</sup> )
Metals	3600	3
Lead	6100	5
Graphite	4830	4.5
Concrete	57000	52
TOTAL	71500	64.5

Table 1. Primary decommissioning waste for total dismantling, RA-1 reactor

# 2.2. Radiological inventory

Table 2 shows contributions of different activated nuclides to the activity after shutdown:

Radionuclide	Half life (year)	Total activity (MBq)
Fe - 55	2.7	$5.52 \times 10^5$
Eu-152	13.6	2.49x10 <sup>5</sup>
Н-3	12.3	2.16x10 <sup>5</sup>
Co-60	5.27	3.33x10 <sup>4</sup>
Ag-110m	0.68	$3.31 \times 10^4$
Mn - 54	0.857	3.26x10 <sup>4</sup>
Eu-154	8.80	$2.68 \times 10^4$
Ni - 63	100	$1.50  ext{ x10}^4$
Cs-134	2.065	$2.32 \times 10^3$
Sb-125	2.76	1.83x10 <sup>3</sup>
Sn-119m	0.803	$1.31 \times 10^3$
Ag-108m	130	$1.17 \times 10^3$
Zn-65	0.67	$0.63  ext{ x10}^3$
Ca - 41	$1.03 \times 10^{5}$	$4.65  ext{ x10}^2$
Cd-109	1.267	$2.82 \times 10^2$
C - 14	5730	$1.48 \text{ x} 10^2$
Ni - 59	$7.50 \mathrm{x10^4}$	$1.25 \text{ x} 10^2$
Sm-151	90	73.8
Eu - 155	4.76	55.7
Cl - 36	3.01 x10 <sup>5</sup>	12.2
Ba-133	10.5	0.31
Total activity		$1.17 \times 10^{6}$

Table 2. Radiological inventory after shutdown

### 2.3. Specific activities

Table 3 shows the calculated average specific activities in the activated components:

Area/Component	Average specific activity (Bq/g)	Mass (kg)	Main nuclides
Control rods	3.2x10 <sup>7</sup>	4	Co-60, Mn-54, Sb-125, Ni-63, Fe-55
Barite concrete	3.1	2528	Eu-152, Ba-133.
Ferrite concrete	$1.07 \times 10^2$	2570	Eu-152, Mn-54, Eu-154, Co-60, Cs-134.
Heavy concrete	3.37x10 <sup>3</sup>	2884	Eu-152, Mn-54, Eu-154, Co-60, Ag-110m.
Lead shielding	$1.02 \times 10^4$	5000	Ag-110m, Sb-125, Ag-108m, Sn-119m, H-3.
Lead shielding thermal column	$1.94 \text{ x} 10^2$	1090	Ag-110m, Sb-125, Ag-108m, Sn-119m.
Graphite thermal column	$9.89  ext{ x10}^3$	2506	Eu-152, Eu-154, Co-60, Mn-54, Zn-65.
Fuel rod grid	6.93 x10 <sup>4</sup>	0.47	Mn-54, Zn-65, Co-60, Fe-55.
Samples pneumatic mechanism	1.47x10 <sup>9</sup>	142	Co-60, Mn-54, Fe-55, Sn-119m, Ni-63.
Basement wall below reactor	8.89 x10 <sup>2</sup>	10245	Eu-152, Eu-154, Co-60, Mn-54, Ag-110m.
Floor around reactor tank	$3.66  ext{ x10}^4$	12540	Eu-152, Eu-154, Co-60, Cs-134, Mn-54.
Floor basement below reactor	3.1x10	19426	Mn-54
Graphite Central reflector	$2.13 \text{ x} 10^4$	20	Zn-65, Mn-54, Co-60, Eu-152, Eu-154.
Graphite outer reflector	$1.11 \text{ x} 10^3$	2000	Co-60, Eu-152, Eu-154, Mn-54, H-3.
Graphite inner reflector	$2.58  ext{ x10}^3$	300	Co-60, Eu-152, Eu-154, Mn-54, H-3.
Reactor tank	$1.77 \text{ x} 10^4$	300	Zn-65, Mn-54, Co-60, Fe-55.
Reactor separating tank	$4.68  ext{ x10}^4$	15,4	Zn-65, Mn-54, Co-60, Fe-55.
Core support structure.	1.00 x10 <sup>5</sup>	0.56	Zn-65, Mn-54, Co-60, Fe-55, Ni-63.

Table 3. Evaluated specific activities of activated materials

Contamination in the reactor room is small and restricted to the cooling circuit well and areas close to the reactor tank; it is considered to be the consequence of a coolant spilling incident in the 70's.

Table 4 shows measured specific activities in a graphite sample from the thermal column:

Radionuclide	Specific activity (Bq/g)
Co-60	5.68
Zn-65	1.40
Ba-133	0.27
Cs-134	0.15
Cs-137	0.25
Eu-152	3.06
C-14	15.8
Cl-36	47.6

Table 4. measured activities in graphite from thermal column

While Table 5 shows some measurements on concrete

Radionuclide	Sample (Bq/g)
Mn-54	1.0
Co-60	9.6
Zn-65	12.9
Cs-137	6.9
Ba-133	1.6
Total	32

# Table 5. Measured activities in concrete close to reactor basement

### 2.4. Metallic waste disposal

Due to the rather low specific activity and small volumes involved, no special problems are likely to show up during disposal of metallic waste.

Decontamination of tanks, piping, valves and pumps will be defined after final shutdown, based on radiological characterization. In the case of lead, self-shielding may make it difficult to measure its radiological characterization. In principle, decontamination of activated components will be avoided in cases it would only increase the amount of radioactive waste.

In the case of most of the piping, pumps and valves, where activation is extremely unlikely, the possibility of decontamination and free-release of materials will be analyzed case by case with the Regulator.

During dismantling of lead shielding, measures must be taken against inhalation of lead dust.

Disposal of metallic waste is planned to be as follows:

In the case of piping, tubes and tanks, either of stainless steel or carbon steel, they will be cut by conventional means and packed and cemented in drums for transport and disposal at the LLW repository. In principle, due to the small volumes involved, compaction does not seem relevant.

Due to their relatively small size, pumps and valves will be packed without cutting. Either cementation and/or backfilling with concrete waste rubble will be used.

Conditioning of aluminium waste is presently under analysis, due to its potential for gas generation. In the RA-1 case aluminium waste is of little importance, in the order of 350 kg, but decommissioning of the other research reactors will add to it.

In the case of lead blocks disposal, consideration must be taken of the repository weight per package acceptance requirements. Determination of the amount of lead in decommissioning waste of other facilities in the country may allow the decision of its use as shielding in waste containers.

#### 2.5. Concrete waste disposal

Due to its relatively low specific activity, disposal of concrete waste does not pose any unique problems. Precautions must be taken with dust generated during the removal of activated

concrete rubble, and may demand the installation of a HEPA-ventilated containment enclosure.

Concrete rubble will be packed in steel drums as low level waste, and transported to the LLW repository site.

Disposal operations will include verification of the conditions of the drums, contact dose determination to fulfil the repository acceptance requirements, and depending on these requirements, setting the drums into the disposal containers, grouting and disposal.

The possibility will be considered of using concrete rubble as backfill for packaging waste consisting of activated/contaminated metallic components, as pumps, valves, etc.

# 2.6. Graphite waste disposal

There is no previous experience in the country on graphite waste management. For this reason, the analysis of graphite disposal included a survey of international experience on the subject.

### 2.6.1. Graphite waste management issues

# (a) Wigner energy

Radiation damage due to fast neutron irradiation of graphite produces large lattice distortions, and energy is stored within the material structure; this stored energy is the so called Wigner energy [2]. When irradiation temperature is high, thermally activated diffusion processes lead to self-annealing of the stored energy. On the other hand, stored energy for low temperature (close to room temperature) irradiations, as in research reactors, can be very high [3], larger than the graphite heat capacity.

Stored energy in irradiated graphite starts to be released when the temperature is increased a few tens of degrees C over the irradiation temperature. In irradiations close to room temperature, the large stored energy, if released, can lead to a positive feedback effect and graphite can reach very high temperatures. This effect may preclude cementation (heat generation during hydration may cause uncontrolled energy releases) and bituminization, unless the stored energy is released before disposal. Also a potential exists for concrete degradation [3, 4].

The complex dependence of Wigner energy on fast neutron flux, neutron spectrum, irradiation time and irradiation temperature makes predictions unreliable, and makes experimental determination (calorimetric methods) the only solution. The same can be said regarding stored energy release.

Several methods have been proposed, and in some cases used, for releasing Wigner energy before storing or disposing of the graphite. It has been shown [5] that heating of room temperature irradiated graphite up to 250°C will release more than 90% of the stored energy.

Incineration of reactor graphite [3] will clearly release the stored energy with the advantage of achieving large volume reductions. The main disadvantages are the dust generated during crashing, the environmental impact of gas releases, and a foreseeable negative impact on the public. An efficient filtration system should exist in an incineration facility in order to deal with usual graphite contaminants as  $Cl^{36}$ ,  $C^{14}$  and Tritium.

New technologies may be developed [6-8]. Promising ones include plasma torch, cold crucible, power laser under gas, a pulsed current method where high intensity brief current pulses are induced in the graphite, leading to its destruction. Isotopic separation technologies aimed to reduce gaseous emissions of  $C^{14}$  may also be developed, and have also been examined in order to reduce  $C^{14}$  from gaseous emissions of reprocessing facilities

In cases where incineration is not performed, safety may demand the release of the stored energy prior to disposal. Convention or radiation heating of graphite blocks is very slow due to the graphite low thermal conductivity (even lowered some order of magnitude by irradiation).

Induction heating [9] is convenient for annealing the stored energy: due to the relatively high electric resistivity of graphite, a penetration of around 5 cm is reached working on the thousands of Hz range. In order to prevent ignition, heating must be performed in an inert gas atmosphere.

# (b) Gas release

Consideration must be taken of eventual activity release while heating irradiated graphite. Reference [5] shows that less of 0.5% of tritium retained in highly irradiated graphite from Windscale Pile 1 was released during heating up to 500°C. Regarding tritium releases, the same reference shows that most of it is as HTO, with more important health implications than elemental HT. The small fraction of tritium released is due to the fact that surface tritium is very easily transferred to the environment, but is only a small fraction of the total content. Further release of bulk tritium is of low significance, due to the low diffusivity of tritium in graphite.

On the other hand, [10] shows high release of  $Cl^{36}$  during graphite heating; this long lived nuclide is usually found, due to neutron activation of impurities containing Cl, as measured in RA-1.

Release of  $C^{14}$  by exchange with gas-phase species will be restricted to its small concentration on the surface (see point j).

### (c) Dust

Both dismantling of nuclear graphite and cutting/milling of graphite generates graphite dust. It is a usual disposal requirement that activity in waste is effectively immobilized and loose particulate material is minimized; in the case of graphite dust, use of a cementitious matrix may be complicated by difficulties in wetting the dust; wetting agents, properly chosen, may then be necessary.

If the dust volumes are small, incorporation in the grout matrix can be achieved [3] and secured by incorporation of a second grout layer above the encapsulation grout. For larger quantities, different alternatives are under analysis, as supercompaction or encapsulation in suitable polymers. The risk of radiolysis of organic compounds, with production of hydrogen in the long term during storage/disposal must be taken into account [8].

# (d) Flotation

Density of irradiated graphite, as low as  $1.6 \text{ g/cm}^3$  in some cases, may be less than density of typical grouts (over  $1.8 \text{ g/cm}^3$ ). This may lead to flotation of waste boxes; consequently some

anti-flotation device may be necessary in the final package. Graphite dust floating is also a possibility, and formation of dust from the graphite bricks should be minimized, and eventually a second grout layer may be necessary.

(e) Galvanic corrosion

The possibility of high corrosion rates exists due to galvanic coupling at graphite – metal contact. However, research carried out in the UK [3] has shown that in anaerobic conditions, galvanic coupling between graphite and steel does not increase significantly the steel corrosion rate. In a disposal package, it may be assumed that corrosion rates during the initial aerobic conditions will fall rapidly as oxygen is consumed by the corrosion process.

### (f) Dust explosion

The possibility of graphite dust explosions has been analyzed, among others, both in the UK [11] and France [6], concluding that the classification of nuclear grade graphites as "non-explosive" is adequate and maintained after irradiation. Extreme tests in the UK nuclear industry have shown that only the presence of a powerful chemical ignition system may lead to graphite explosion [12].

(g) Ignition

In order to produce flames or fire propagation in graphite, the following conditions must be satisfied: high surface area to volume ratio, temperature over 650°C, adequate oxygen supply but not sufficient gas flow to cause cooling, high intrinsic reaction rate and low heat loses [12]. In this sense, it may be interesting to note that later analysis has shown that it was not the graphite but the fuel which ignited during the well known Windscale Pile 1 fire.

#### (h) Oxidation

The reaction rate for oxidation of nuclear graphite in air is very low. Even assuming the presence of inorganic catalysts due to leaching from concrete or metallic structures, the reaction rate remains small, and may be eventually controlled by the addition of oxidation inhibitors, where experience exists from other uses of carbon. In order to achieve direct reaction between carbon and water, extremely high temperatures (over 900°C) are required even in the presence of catalysts. This implies that graphite oxidation is not an issue when analyzing storage or disposal options.

#### (i) Leaching

Relative leach rates of most usual radionuclides present in irradiated graphite are [6]:

$$Cs^{137} > Ba^{133} > Co^{60} = Ni^{63} > Cs^{134} > Cl^{36} > Eu^{154} > H^3 > C^{14}$$

This reference provides information on research done in the US, France, the UK and the Russian Federation. Reference [13] shows that in irradiated graphite, leaching is restricted to  $C^{14}$  on the graphite surface, which in turn is mainly due to surface activation of N<sup>14</sup>. The reference also shows the in-depth distribution of N in graphite, consistent with previous results that take into account the extremely small diffusivity of carbon atoms in graphite.

#### 2.6.2. Disposal of graphite from dismantling of Argentinean research reactors

Graphite waste disposal in Argentina will not be restricted to material resulting from dismantling the RA-1 reactor. Graphite reflectors also exist in the 10 MW radioisotope production reactor RA-3, in the Centro Atómico Ezeiza close to Buenos Aires city, and in the 500 kW research reactor RA-6 in the Centro Atómico Bariloche, in the south-west of the country. Nevertheless, graphite waste disposal in Argentina is in the order of less than 100 metric tons.

Because the three graphite-containing research reactors are currently operating, experimental determination of radionuclide inventory and eventual Wigner energy is difficult, and is only obtained during maintenance operations, as has been the case with RA-1. Therefore, radiological characterization will be based mainly on computer modelling, as has been already performed for RA-1 and will be performed in the future with RA-3 and RA-6. Computer modelling will also become an important tool for reducing the number of samples necessary to evaluate the radionuclide inventory and stored energy once the reactors are permanently shutdown.

Graphite reflectors in the Argentinean research reactors operate at room temperature, and build-up of Wigner energy is a possibility.

After shutdown of a research reactor, the first step regarding graphite waste management will be measurement of stored energy, by adequate sampling helped by activation calculations. This may be done by calorimetric methods, a technique existing in CNEA laboratories.

For those cases in which the amount of stored energy may imply safety risks, annealing will become necessary. As shown before, induction heating with control of gaseous emissions is an adequate technique, and experience also exists in CNEA regarding this kind of technology.

The rather small graphite waste volumes do not justify the costs and resources necessary to develop a volume-reducing technology, unlike the several incineration alternatives and gaseous emissions treatments described in the previous section. Consequently, after dealing with stored energy, graphite blocks will be packed in containers and transported to a storage or disposal facility. As a way of minimizing graphite breakage and dust generation, graphite blocks may be cemented or grouted inside the drums. Small parts and dust coming from breakage, etc., will be cemented before packing in containers.

Figure 1 shows a schematic of the proposed graphite waste management, with the following tasks:

- Dismantling: graphite reflectors are dismantled, taking care of minimizing the amount of dust generated. Dust and broken parts are collected.
- Sampling: based on previous measurements and on the computer modelling, a sampling strategy is defined. There will be two kinds of samples: for radionuclide inventory determination, and for stored energy measurement.
- Based on the stored energy measurements and using the computer modelling both for interpolation and extrapolation, graphite blocks are classified depending on whether stored energy must or must not be annealed. It is considered that most of the graphite from the central graphite and first layer of the internal annular reflector will demand annealing, in order to avoid incidental releases of energy with deleterious effects on container backfilling.

- Annealing in an induction furnace under inert atmosphere is performed. Some secondary waste may be collected in the HEPA filters, mainly Cl<sup>36</sup>. Secondary waste is treated and packaged.
- Once acceptance requirements for the LLW and ILW repository are defined, the detailed packaging procedure will be defined. This includes cementation or grouting of graphite in the container.
- Graphite waste is then transported to the repository site.
- Disposal operations include verification of the conditions of the boxes, contact dose determination to fulfil the repository acceptance requirements, putting the graphite boxes into the disposal containers, grouting and disposal.
- 3. CONCLUSIONS

The following conclusions can be established:

- Most of the RA-1 decommissioning waste will be low level concrete rubble, whose disposal should not pose any particular challenge.
- Metal waste, aluminium and steel, due to its rather low specific activity, should not pose particular disposal problems as, i.e., gas generation.
- Graphite waste from reactor reflectors, due to lack of previous experience in the country with this type of waste, will demand special consideration. Stored energy should be determined, and in some cases induction annealing may become necessary before waste conditioning and disposal.
- Participation in the CRP has resulted in an experience and procedures that will be applied to decommissioning planning, waste prediction and decommissioning waste disposal analysis for the other 5 research reactors in the country.

Future activities will incorporate lessons learned from this CRP:

- Disposal aspects of decommissioning waste must be considered in the definition of acceptance requirements for the national Low Level Repository and Intermediate Level Repository. If necessary, some of the proposed management strategies should be reviewed.
- Even if the amount of graphite waste from total dismantling of all research reactors in the country will be small (less than one hundred tons), presence of long-lived nuclides as C<sup>14</sup> and Cl<sup>36</sup> imposes the convenience of improving the characterization of graphite, including determination of stored energy and eventual need for annealing. This would simplify the situation in case premature shut down of some facility occurs.
- Decommissioning waste prediction can be improved if a radiological characterization plan based on samples taken during reactor maintenance shut downs is implemented.
- Calculation of materials activation can be improved by using Monte Carlo codes.
- Quantitative Neutron Activation Analysis may be the more adequate technique for measuring materials composition at the trace (ppm) level.
- Selection of materials (including concrete aggregates) minimizing the concentration of impurities leading to long-lived nuclides by neutron activation (i.e. Eu, Ho, Cl, Li) may be an efficient method for lowering both decommissioning waste and occupational doses [14].



FIG. 1 Graphite waste management.

#### REFERENCES

- [1] VETERE. C., EPPENSTEIN, M., Networking for capturing and preserving existing Knowledge. IAEA Technical Meeting on Managing Nuclear Knowledge, ICTP, Trieste, Italy, August 2005.
- [2] INTERNATIONAL ATOMIC ENERGY AGENCY, Irradiation Damage in Graphite Due to Fast Neutrons in Fission and Fusion Systems, IAEA-TECDOC-1154, Vienna (2000).
- [3] WISE, M., Management of UKAEA graphite liabilities. IAEA Technical Committee Meeting on Nuclear Graphite Waste Management, Manchester, UK (1999).
- [4] GUPPY, R.M., McCARTHY, J., WISBEY, S.J., Technical assessment of the significance of Wigner energy for disposal of graphite wastes from the Windscale piles. In IAEA Technical Committee Meeting on Nuclear Graphite Waste Management, Manchester, UK (1999).
- [5] WÖRNER, J., BOTZEM, W., PRESTON, S.D., Heat treatment of graphite and resulting tritium emissions. IAEA Technical Committee Meeting on Nuclear Graphite Waste Management, Manchester, UK (1999).
- [6] GUIROY, J.J., Graphite waste incineration in a fluidized bed. IAEA Specialist Meeting on Graphite Reactor Lifecycle Behaviour. Bath, UK (1995).
- [7] SAISHU, S., INOUE, T., Technical development of graphite waste treatment in NUPEC. Technical Committee Meeting on Nuclear Graphite Waste Management, Manchester, UK (1999).

- [8] DELAGE, F., LATGE, C., THOUVENOT, P., Overview of graphite treatment processes in High Temperature Reactors, 2<sup>nd</sup> Intern. Topical Meeting on High Temperature Reactor Technology, Beijing, China, Sept. (2004).
- [9] BOTZEM, W., WÖRNER, J., Inert annealing of irradiated graphite by inductive heating. IAEA Technical Committee Meeting on Nuclear Graphite Waste Management, Manchester, UK (1999).
- [10] BROWN, F.J., PALMER, J.D., WOOD, P., Derivation of a radionuclide inventory for irradiated graphite. Chlorine-36 inventory determination. IAEA Technical Committee Meeting on Nuclear Graphite Waste Management, Manchester, UK (1999).
- [11] MARSDEN, B.L., PRESTON, S.D., WICKHAM, A.J., TYSON, A., "Evaluation of graphite safety issues for the British production piles at Windscale: graphite sampling in preparation for the dismantling of pile 1 and the further safe storage of pile 2", in Technologies for Gas Cooled Reactor Decommissioning, Fuel Storage and Waste Disposal, IAEA-TECDOC-1043, IAEA, Vienna (1998) p. 213.
- [12] WICKHAM, A.J., MARSDEN, B.J., SELLERS, R.M., PILKINGTON, N.J., "Graphite core stability during "care and maintenance" and "safe storage", Technologies for Gas Cooled Reactor Decommissioning, Fuel Storage and Waste Disposal, IAEA-TECDOC-1043, IAEA, Vienna, (1998) p. 225.
- [13] TAKAHASHI, R., TOYAHARA, M., MARUKI, S., UEDA, H., Investigation of morphology and impurity of nuclear grade graphite, and leaching mechanism of Carbon-14". IAEA Technical Committee Meeting on Nuclear Graphite Waste Management, Manchester, UK (1999).
- [14] KINNO, M., KIMURA, K., NAKAMURA, T., Raw materials for low-activation Concrete Neutron Shields". J.Nucl. Sci. and Technol. Vol.39, No.12, p.1275–1289, Dec. 2002.

#### CANADA

#### DECOMMISSIONING OF ONTARIO POWER GENERATION'S NUCLEAR STATIONS — RADIOACTIVE WASTE VOLUME ARISINGS AND RADIONUCLIDE INVENTORIES

#### A. Husain

Kinectrics, Inc.

#### Abstract

Ontario Power Generation (OPG) is a government owned electrical utility operating in the province of Ontario, Canada. It owns five 4-Unit Nuclear Generating Stations. For planning purposes, OPG stations are assumed to be shut down after 40 years of operation and then decommissioned in accordance with a delayed dismantling strategy. An overall analysis, considering both waste volume arisings and radionuclide inventories, was performed to develop a reference database on the inventory and characteristics of potential Low and Intermediate Level Waste arisings from decommissioning. This information, in turn, has contributed towards the development of a reference disposal plan for the decommissioning waste.

An overall analysis was performed to develop a reference database for the inventory and characteristics of potential Low and Intermediate Level wastes from the decommissioning of OPG stations. This information, in turn, will contribute towards the development of a reference disposal plan for the decommissioning waste.

Previous papers presented an analysis of waste volume data for Darlington NGS (presented at the 1st CRP Meeting in Sellafield) and an overview of the radionuclide inventory assessment for activated decommissioning waste from Darlington NGS (presented at the 2nd CRP Meeting in Buenos Aires). A third paper dealing with the integrated findings on waste volume data for all OPG stations has been submitted to the IAEA as part requirement for the 3rd and final CRP meeting in Vienna. This present paper represents an overall summary of the findings reported in all the three papers.

#### 1. INTRODUCTION AND BACKGROUND

Ontario Power Generation (OPG) is a government owned electrical utility operating in the province of Ontario, Canada. It owns five 4-Unit Nuclear Generating Stations (NGS). These are located on three sites, namely, Pickering (Stations A and B), Bruce (Stations A and B) and Darlington. Although all the plants are based on the CANDU reactor concept, key differences exist between their designs.

For long term planning purposes, it is assumed that OPG stations will be shut down following their 40-year operating lives and then decommissioned. A delayed dismantling strategy is assumed. The strategy is characterized by the following three stages:

- Stage 1 Preparation for Safe Storage:
- Stage 2 Safe Storage (duration 30 years)
- Stage 3 Preparation for Dismantling, Dismantling and Site Restoration

For the purpose of developing decommissioning plans, TLG Services, Inc. were contracted by OPG in 1998-2000 to prepare cost estimates for decommissioning Bruce, Pickering and Darlington stations. Based on a breakdown of work activities involved in the three

decommissioning stages, the methodology employed by TLG also generated a description of the decommissioning waste arisings and their volumes after processing and packaging.

The TLG data were subsequently examined by Kinectrics and led to a detailed analysis of the waste arisings from each station. An overall waste analysis was required to develop a reference database for the inventory and characteristics of potential Low Level Waste (LLW) and Intermediate Level Waste (ILW) arisings from decommissioning. This information, in turn, contributed to the development of a reference disposal plan for the decommissioning waste. Also, a separate Kinectrics study in 2001-2 focused on the detailed radionuclide inventories associated with Darlington's activated decommissioning waste; similar studies have been undertaken in the mid 80's for Pickering and Bruce generating stations.

Based on Kinectrics' work in support of OPG's decommissioning plan development, two papers were previously presented at meetings of the Coordinated Research Project (CRP), namely,

- Paper I (presented at the 1st Meeting in Sellafield) dealt with an analysis of waste volume data for Darlington NGS.
- Paper II (presented at the 2nd Meeting in Buenos Aires) presented an overview of the radionuclide inventory assessment of activated decommissioning waste from Darlington NGS.

A third paper dealing with the integrated findings on waste volume data for all OPG stations has been submitted to the IAEA as part requirement for participation in the CRP. This present paper represents an overall summary of the findings reported in all the three papers. The reader is referred to the earlier papers for background information pertaining to the CANDU reactor system and systems and structures at OPG's generating stations.

# 2. METHODOLOGY EMPLOYED TO ESTIMATE DECOMMISSIONING WASTE VOLUME ARISINGS

TLG developed an inventory of installed components and structures at each station based on drawings, document reviews and on-site surveys. Their inventory data were reviewed to gather details relating to radioactive waste. Data on waste volumes were organized into two distinct groups:

Group 1 was based on the radioactive characteristic of the wastes:

- Activated Wastes This waste consists of reactor components and systems that are in close proximity to the reactor core and become radioactive as a result of neutron irradiation. The induced radioactivity is distributed throughout the irradiated volume.
- Primary Contaminated Wastes This refers to outcore system components which become contaminated during operation as a result of the transport and deposition of activation and fission products.
- Secondary Contaminated Wastes This refers to waste generated during various decontamination and dismantling activities and consists primarily of DAW. Also included are solid wastes resulting from periodic surveys of the dormant facility (Stage 2), from maintenance activities and from treatment of liquids collected on-site.

Group 2 was based on the manner of waste processing:

- Non-Processible (NP) Wastes This category includes all radioactive structural materials (e.g. concrete) and metal components which are not amenable to processing or treatment (beyond the initial decontamination treatment);
- Processed Liquids This category refers to solid wastes resulting from the treatment of liquid wastes by processes such as ion exchange and filtration. The spent media are assumed to be conditioned in a cementitious matrix. Also, included in this category are conditioned evaporator concentrates.
- Compacted Wastes This category represents dry active waste (DAW) consisting of paper, cloth, wood, plastics, etc. The waste is amenable to volume reduction by medium force compaction.

TLG's estimates of decommissioning waste arisings are based on several specific assumptions. These have been detailed in previous papers prepared for the CRP.

#### 3. DECOMMISSIONING WASTE ARISINGS

#### 3.1. Overall Waste Arisings

Waste volume arisings from the decommissioning of OPG's nuclear stations are summarized in Table 1. The volumes represent the total of containerized waste and the displacement volumes of intact items. The data are based on TLG's 1998-2000 reports. However, the estimates have been revised recently but will not be available until a draft 2005 TLG report prepared for OPG is finalized later in 2006. Based on the data in Table 1, LLW constitutes 88 % of the overall decommissioning waste.

Station	Rated	Packaged V	Packaged Volume (m <sup>3</sup> )	
	(MWe)	LLW	ILW	
PNGS-A	515	20,172	2,041	
PNGS-B	515	14,294	2,018	
BNGS-A	769	15,694	2,528	
BNGS-B	785	16,763	2,527	
DNGS	881	25,417	3131	
	All	92,340	12,245	

#### Table 1. Estimated decommissioning waste inventories for OPG stations

Some of the differences in waste volumes between the stations are attributable to their design differences although the varying quality of inputs (plant drawings, databases and documentation) available for the three TLG studies is probably an important factor also: the BNGS study was the first to be undertaken while the DNGS study was the last, with the available information in the case of DNGS being the most comprehensive.

Part of the difference in the LLW estimates for PNGS-A and PNGS-B are attributed to:

 PNGS-A calandria has a greater mass than the PNGS-B calandria because of shielding design differences. The former also houses ten additional pressure tubes.

- Unlike PNGS-B, PNGS-A utilizes a moderator dump tank (for fast shutdown) which is located directly below the calandria.
- Because of differences in vault design, the surface of the PNGS-A reactor vault concrete would be removed to a depth of at least 18 inches, thus contributing to additional LLW.

The somewhat higher LLW estimate for BNGS-B compared with that for BNGS-A are attributed to the following:

- --- Steam generators at BNGS-B have individual steam drums while those at BNGS-A have a common steam drum, and
- --- Moderator heat exchangers at BNGS-B have a larger volume than those at BNGS-A.

The following sections present further analysis of the waste arisings data.

### 3.2. Description of Waste Arisings

Radioactive waste generated during Stage 1 includes low level compacted DAW and intermediate level treated process liquid wastes. That generated during Stage 2 consists primarily of low level compacted DAW from periodic surveys and facility inspections. Waste arisings in Stage 3 consists primarily of dismantled system components and structural materials including:

- calandria components such as pressure tubes, calandria tubes, end fittings and reactor control mechanisms, and shield balls (these are classified as ILW),
- large intact metallic components such as steam generators and pressurizers (these are classified as LLW),
- process system components such as piping, valves and pumps (these are classified as LLW), and
- contaminated concrete associated with containment, active drainage areas and fuel bays (these are classified as LLW).

Waste arisings data (based on the 1998-2000 TLG reports) from OPG stations during their various decommissioning stages are depicted in Figures 1-3. The dates shown were consistent with the planning assumptions which were current at the time the 1998-2000 TLG studies were undertaken. These dates, however, do not correspond to current OPG plans; for instance, Bruce A is currently planned for operation until 2036 and two of the units at PNGS-A are planned to operate until 2027. Planned refurbishment may extend the decommissioning start dates for various units beyond those considered. Thus, the time scales shown in Figures 1-3 should be considered merely for illustrative purposes.

Key points based on Figures 1-3 are summarized below:

- A major portion (~97%) of the total LLW arisings is generated during Stage 3. The ILW volume generated during Stage 1 is significant and represents about 42% of the total ILW arisings.
- NP waste is the dominant category because current planning assumes only DAW to be compacted. NP wastes are generated only in the dismantling stage. Approximately 83 %

of the NP waste is composed of metal, the balance being concrete. Most of the NP waste ( $\sim$ 93 %) is LLW.

Processed liquid waste falls into the ILW class and represents 50 % of the total ILW.

Overall, contaminated waste (both primary and secondary) represents 64% of the total LLW and ILW, with the balance being activated waste. 'Segmented Calandria', 'Calandria Internals' and 'Concrete' represent the three major activated waste streams. 'Calandria Internals' represents the only activated ILW stream. Spent fuel storage frames and steam generators are two of the largest waste streams in the contaminated waste category. All activated waste, 91% of the primary contaminated waste and 74% of the secondary contaminated waste are generated during Stage 3.



Fig. 1. Overall waste arisings during various decommissioning stages



Fig. 2. Arisings of NP waste, compacted waste and processed liquids during various decommissioning stages



Fig. 3. Arisings of activated and contaminated waste during various decommissioning stages

# 3.3. Waste Package Arisings

Most of the LLW, except components which would be disposed of intact, would be packaged in standard steel containers. All ILW will be packaged in liners for transportation in shielded casks. Table 2 presents estimates for the number of containers required to dispose OPG's decommissioning waste; the estimates are based on TLG's 1998-2000 waste arisings data. Approximately 25 926 LLW and 5236 ILW containers (total of 31 162 containers) would be needed. In contrast, the number of intact waste packages was estimated to be 1092 with a corresponding volume of 28 571 m<sup>3</sup>.

Packages	PNGS-A	PNGS-B	BNGS-A	BNGS-B	DNGS	All
LLW containers	4,513	3,456	5,790	5,793	6,374	25,926
ILW containers	840	780	1,215	1,215	1,186	5,236
All containers	5,353	4,236	7,004	7,008	7,560	31,162
Intact Items	423	354	52	52	211	1092

Table 2.	<b>Estimates of</b>	waste container	requirements f	or OPG's d	lecommissioning v	vaste
			1		8	

Steam generators and pressurizers represent the largest components which were originally planned to be disposed of intact. These would have to be individually transported. The largest steam generator has a volume of 250 m<sup>3</sup> compared with a gross volume of 15 m<sup>3</sup> for the largest container planned to be used for the decommissioning waste. Based on the size and weight restrictions of the deep geological repository (DGR) currently being planned by OPG, large components such as steam generators are now planned to be cut up and packaged instead of being disposed of intact.

Considering the restrictions imposed by the DGR design, the total number of LLW containers will be significantly greater than the estimates in Table 2. This increase will be offset by a somewhat lower number of intact components.

### 4. INVENTORY OF RADIONUCLIDES IN DECOMMISSIONING WASTE

OPG reports prepared in the mid 1980s present estimates for the overall radionuclide inventories in Pickering and Bruce decommissioning waste based on ORIGEN code calculations for various reactor core components. Similar estimates for Darlington were prepared by Kinectrics during 2001-2. The overall methodology and results obtained for Darlington were presented during the course of the second CRP at Buenos Aires. A brief account of this work and the findings are presented below.

Table 3 presents a breakdown of the various irradiated reactor components at Darlington and their assumed waste classification when the reactor is decommissioned. Estimates of radionuclide activities associated with the various components listed in Table 3 were developed considering only neutron-induced activation. Activity from deposited activation products is implicitly accounted for in these estimates; compared with them, deposited fission product activities are expected to be relatively small. Also, radionuclides deposited on system surfaces, in general, are likely to be partially removed from the reactor system as a result of

decontaminations performed during the operating life of the reactor. This would further reduce the inventory of deposited radionuclides in decommissioning waste.

	Quantity	Mass (lb)	Assumed
Component	Per Unit	per	Type of
		Component	Waste*
Shield Tank-Calandria-End Shields Assembly			
Shield Tank (less extension)	1	689,000	LLW
Calandria vessel	1	83,990	LLW
Calandria tubes with inserts	480	54	ILW
Internal piping (shield tank to calandria)	1	15,800	LLW
End Shields	2	169,570	LLW
Shield Tank Extension	1	69,970	LLW
Steel Balls			
Shield Tank	1	180,500	LLW
Shield tank extension	1	142,700	LLW
End shields	2	283,400	ILW
Solid Shielding – Shield Tank			
Curtain shielding slabs	2	27,531	LLW
Annular shielding assemblies	72	2,820	LLW
Fuel Channels		· · ·	
End fitting assemblies	960	379	ILW
Closure plugs	960	28	LLW
Shield plugs	960	173	ILW
Liner and latch	960	70	LLW
Pressure tubes	480	138	ILW
Grayloc fittings	960	18.9	LLW
Reactivity Mechanism Deck			
Deck structure	1	127.166	LLW
Vertical shielding	1	43.206	LLW
Concrete shielding	1	152,895	LLW
Tread plates and floor plates	1	59,527	LLW
Manhole plug	1	1,174	LLW
Shielding collars	1	16,897	LLW
ZCU <sup>b</sup> shielding box and support	2	22,687	LLW
ZCU <sup>b</sup> horizontal shielding slab	1	21,466	LLW
Other		,	
Control rods with drives and ion chambers	1	128.000	ILW
Feeder tubes	1	388.875	LLW
External piping - end shield cooling (full)	2	6.500	LLW
- shield tank cooling	1	2,000	LLW
Feeder tube supports and hangars	1	8,600	LLW
Feeder cabinet	2	2,000	LLW
Gap shielding tank extension to concrete	1	37,000	LLW
Shield tank support bearings incl. base plate	4	16,400	LLW
		·	LLW

# Table 3. Mass and waste classification of Darlington NGS reactor components

\* Assumed waste type for planning purposes; LLW, Low Level Waste; ILW, Intermediate Level Waste

Windows-based ORIGEN 2 code<sup>1</sup> was used to perform neutron activation calculations. The programme inputs included (a) power history, (b) neutron flux, (c) detailed material composition (d) mass of irradiated component (e) temperature and (f) decay period. Calculations were performed considering 40 years of operation at 85 % power followed by 33 years of decay during storage in Stages 1 and 2.

### 4.1. Estimates of Total Radionuclide Activities for Various Darlington In-core Components

Results obtained for all components are summarized in Tables 4 and 5, based on which the following conclusions were drawn:

The overall activity was estimated to be 4.2E+16 Bq. End shields and the calandria vessel, respectively, account for approximately 60 % and 33 % of the overall activity.

- The overall activity is dominated (~95%) by Ni-63. Other radionuclides present, in order of decreasing importance, are Co-60, {Ni-59, Nb-94, C-14 and Fe-55}, Zr-93, Mo-93 and Cl-36 (activities of the bracketed radionuclides are essentially similar). The end shields are the dominant source for C-14, Fe-55, Co-60, Ni-59, Ni-63 and Cl-36 activities.
- The pressure tubes are the principal source of Zr-93 and Nb-94. Next to pressure tubes, calandria tubes are a significant source for Zr-93. The annular shielding assembly is the principal source for Mo-93.

### 4.2. Comparison of Radionuclide Inventories for Various OPG Stations

Table 6 presents a comparison between the inventory estimates for Darlington obtained in this study with the corresponding estimates for Pickering NGS-A and Bruce NGS-A reactor unit. The Pickering and Bruce calculations considered 40 years of reactor operation followed by 30 years of storage.

With the exception of Nb-94, estimated radionuclide activities in Darlington NGS decommissioning waste are essentially similar to those previously estimated for Pickering NGS-A and Bruce NGS-A stations. The estimated level of Nb-94 in Darlington NGS decommissioning waste appears to be a factor of 29 greater than the corresponding level at Bruce NGS-A, which in turn appears to be a factor of 19 greater than the level in Pickering NGS-A waste. Considering that pressure tubes are the principal source of Nb-94 and that each Pickering, Bruce and Darlington reactor has 390, 480 and 480 pressure tubes, respectively, with each pressure tube being of approximately similar mass, it is not evident why the Nb-94 levels at the three stations differ so significantly.

<sup>&</sup>lt;sup>1</sup> Simplified Neutron Activation Analysis Program (SNAP) developed by D.W. James & Associates.

Reactor					Radion	uclide Activ	rity (Bq)				
Components	C-14	Fe-55	Co-60	Ni-59	Ni-63	Zr-93	Nb-94	CI-36	Mo-93	Other	AII
ndria Vessel	3.3E+13	2.1E+13	2.5E+14	1.3E+14	1.3E+16			6.4E+06			1.4E+16
Indria Tubes with Inserts	1.1E+13	2.8E+12	6.3E+12	8.8E+11	1.6E+14	2.7E+12	2.0E+07	9.8E+06	4.4E+09	4.2E+12	1.8E+14
rnal Piping	2.9E+10	1.9E+10	2.4E+11	1.4E+11	1.2E+13						1.3E+13
shields	1.2E+14	4.1E+13	8.2E+14	2.0E+14	2.4E+16			2.2E+07		1.9E+07	2.5E+16
eld Tank Steel Balls	1.0E-02	6.8E+02	7.5E+02								1.4E+03
Shield Steel Balls	3.9E+08	2.4E+13	6.9E+13	7.3E+05				5.2E+06		1.4E+06	9.4E+13
ain Shield Slabs	5.2E-04	1.1E+01	4.2E+01						5.0E-02	7.0E-03	5.3E+01
ular Shielding Assembly	2.4E+08	5.0E+12	2.9E+13				5.4E+06		2.8E+10	6.2E+09	3.4E+13
Fitting Assemblies	1.8E+08	9.4E+12	2.5E+13	1.6E+12	1.5E+14		2.4E+06	4.5E+05	1.4E+10	1.8E+09	1.8E+14
ure Plugs	1.4E+04	3.0E+05	8.1E+05	1.9E+05	1.7E+07	1.2E-04	1.4E-02		7.2E+01	2.0E+01	1.8E+07
Id Plugs	3.3E+05	2.4E+10	9.1E+10	3.1E+10	2.8E+12	1.1E+03	1.3E+05	6.3E+00	7.0E+08	1.5E+08	2.9E+12
rs & Latches	1.3E+06	2.1E+11	4.4E+09	4.9E+10	4.4E+12					4.5E+09	4.7E+12
sure Tubes	2.0E+13	1.4E+11	9.2E+12	7.5E+10	1.6E+13	7.6E+12	2.9E+14		1.2E+10	1.1E+12	3.4E+14
er Springs	6.8E+09	9.6E+09	2.9E+11	1.0E+12	2.2E+14	2.3E+09	1.6E+04	4.0E+04	1.5E+06	4.9E+09	2.2E+14
/loc Fittings	9.6E+00	6.4E+05	9.2E+05			1.3E-02	1.6E+00		8.2E+03	1.8E+03	1.6E+06
ing Sleeves	2.0E+08	4.2E+12	1.7E+13		9.8E+08	4.5E+04	4.1E+06		2.4E+10	4.2E+09	2.2E+13
ctor Control Mechanisms	2.4E+13	6.8E+12	4.9E+13	9.7E+12	2.3E+15	1.2E+12	1.0E+07	1.1E+07	8.0E+08	7.1E+12	2.4E+15
	2.1E+14	1.2E+14	1.3E+15	3.5E+14	4.0E+16	1.2E+13	2.9E+14	5.4E+07	8.5E+10	1.2E+13	4.2E+16

Table 4. Summary of Radionuclide Inventory Data - Contribution of Key Radionuclides

Reactor Components				Per	centage Distribut	ion			
	C-14	Fe-55	Co-60	Ni-59	Ni-63	Zr-93	Nb-94	CI-36	Mo-93
Calandria Vessel	1.6E+01	1.8E+01	2.0E+01	3.7E+01	3.3E+01			1.2E+01	
Calandria Tubes with Inserts	5.4E+00	2.4E+00	4.9E-01	2.5E-01	3.9E-01	2.3E+01	90-30 <sup>.</sup> 7	1.8E+01	5.2E+00
Internal Piping	1.4E-02	1.6E-02	1.9E-02	4.0E-02	3.1E-02				
End shields	5.8E+01	3.6E+01	6.4E+01	5.9E+01	6.0E+01			4.0E+01	
Shield Tank Steel Balls	4.9E-15	5.9E-10	5.9E-11						
End Shield Steel Balls	1.9E-04	2.1E+01	5.4E+00	2.1E-07				9.5E+00	
Curtain Shield Slabs	2.5E-16	9.9E-12	3.3E-12						5.9E-11
Annular Shielding Assembly	1.1E-04	4.3E+00	2.3E+00				1.9E-06		3.3E+01
End Fitting Assemblies	8.6E-05	8.1E+00	2.0E+00	4.6E-01	3.6E-01		8.4E-07	8.3E-01	1.7E+01
Closure Plugs	6.5E-09	2.6E-07	6.4E-08	5.5E-08	4.3E-08	1.0E-15	4.8E-15		8.5E-08
Shield Plugs	1.6E-07	2.1E-02	7.2E-03	9.0E-03	6.9E-03	9.8E-09	4.7E-08	1.2E-05	8.3E-01
Liners & Latches	6.4E-07	1.8E-01	3.4E-04	1.4E-02	1.1E-02				
Pressure Tubes	9.8E+00	1.2E-01	7.2E-01	2.1E-02	4.0E-02	6.6E+01	1.0E+02		1.4E+01
Garter Springs	3.3E-03	8.3E-03	2.2E-02	2.9E-01	5.5E-01	1.9E-02	5.7E-09	7.4E-02	1.7E-03
Grayloc Fittings	4.6E-12	5.6E-07	7.3E-08			1.1E-13	5.4E-13		9.6E-06
Bearing Sleeves	9.6E-05	3.6E+00	1.4E+00		2.4E-06	3.8E-07	1.4E-06		2.8E+01
Reactor Control Mechanisms	1.1E+01	5.9E+00	3.9E+00	2.8E+00	5.8E+00	1.0E+01	3.6E-06	1.9E+01	9.4E-01
Total	1.0E+02	1.0E+02	1.0E+02	1.0E+02	1.0E+02	1.0E+02	1.0E+02	1.0E+02	1.0E+02

Table 5. Summary of Radionuclide Inventory Data - Percentage Distribution of Key Radionuclides

Dadianualida	commissioning		
Kaulonuchue	Pickering NGS- A	Bruce NGS- A	Darlington NGS (Present)
C-14	1.8E+14	1.4E+14	2.1E+14
Fe-55	9.5E+13	1.6E+14	1.1E+14
Ni-59	3.8E+14	3.8E+14	3.5E+14
Co-60	3.0E+15	1.1E+15	1.3E+15
Ni-63	2.8E+15	3.8E+16	4.0E+16
Zr-93	1.6E+13	1.1E+13	1.2E+13
Nb-94	5.3E+11	1.0E+13	2.9E+14

 Table 6. Comparison between inventory estimates for key activation radionuclides in

 Pickering NGS-A, Bruce NGS-A and Darlington NGS decommissioning wastes

# 5. CONCLUSIONS

A detailed work breakdown structure for decommissioning activities combined with credible assumptions for management of the resulting waste has provided a detailed understanding of waste volumes, types of waste and the chronology of waste generation for OPG's decommissioning waste. The waste differs significantly in composition from operational waste; in particular, it is characterized by a relatively high metal content.

In addition to the detailed estimates for decommissioning waste volume arisings, activation calculations have provided a through understanding of the quantities and distribution of various long-lived radionuclides present in the waste.

The current estimates provide a sound basis for decommissioning planning, decommissioning cost estimations and repository design. The assessments presented here may be periodically updated in future based on revised planning scenarios and considering advances in decommissioning technologies.

### ACKNOWLEDGEMENTS

This work was funded by the Nuclear Waste Management Division of Ontario Power Generation. The author is grateful for comments received from M. Garamszeghy, N. Jayawardene and J. Krasznai.

#### CHINA

#### FEASIBILITY STUDY OF THE OPTION OF ON-SITE DISPOSAL OF DECOMMISSIONED NUCLEAR WASTES IN CHINA

#### Z. Wen, Z. Liu

Beijing Research Institute of Uranium Geology

#### Abstract

On-site disposal, which implies that most of the radioactive material will be permanently emplaced in an engineered structure near the surface, may sometimes be a reasonable option for the specific decommission waste that is characterized by large volume and very low-level activity. At the ER site, an on-site disposal facility was constructed to dispose of low-level operational wastes and contaminated soil and demolition debris during decommissioning process. The disposal pit was constructed by excavation. The depth of the disposal pit is about 7 meters. In order to retard the nuclides to escape from disposal pit into groundwater, the bottom of the disposal pit was filled with 50cm compacted local red soils and the side walls were filled with 30cm compacted clay. The drainage ditches are constructed to discharge rainwater and the surface water. The slope of the ground surface in the disposal zone is  $3^{\circ} \sim 5^{\circ}$ in length and 10°~15° in breadth, which favours the rainfall flowing to drainage ditches rather than advancing by infiltration. The wastes are disposed of layer by layer. The thickness of each layer about 20cm and there are 25 layers in total. In order to reduce potential erosion, bio-intrusion, and seepage, a multiple layer capping is designed, which is composed of 8 sublayers. The inspection and verification shows that the engineering measures can successfully assure the isolation of decommissioned wastes from the surrounding environment. The gamma survey shows that the penetration radiation rate on the surface of the cover is just the similar to that in the surrounding environment. A set of analysis of samples from the surrounding environment hasn't shown the migration of the uranium from the disposal pit to the environment, which proves the effectiveness of the on-site disposal facility.

#### Key words: On-site disposal, Decommissioning waste, China

#### 1. INTRODUCTION

In most cases, decommissioning of nuclear facilities is accomplished under two basic strategies, namely: (1) immediate dismantling; or (2) safe enclosure followed by deferred dismantling. Both strategies are intended to lead eventually to unrestricted release of the site and imply removal of radioactive waste to an off-site repository. There is however a third strategy called on-site disposal, which consists of disposing of the nuclear facility on the same site where it had operated. Variations exist, ranging from local disposal of some waste to disposal of complete nuclear facilities such as reactor plants and fuel cycle facilities (IAEA, 1999).

On-site disposal, which implies that most of the radioactive material will be permanently emplaced in an engineered structure near the surface, sometimes nay be a reasonable option for the specific decommission waste that is characterized by large volume and very low-level activity. This report presents the feasibility study of the option of on-site disposal of decommissioning wastes in northwest China.

# 2. LAWS, REGULATIONS AND FRAMEWORK OF STANDARDS FOR RADIOACTIVE WASTE DISPOSAL

At the beginning of 1980s, radwaste management was focused on the disposal of waste. Since then some policies, programmes, standards and technical criteria have been developed to meet the requests of radwaste disposal (Wang Xiande, 1997). The Policy and Principles on Disposal of Low-and Intermediate-level Radioactive Solid Waste was promulgated in 1992 and approved by the State Council, dealing with the matters in disposal of solid low-and intermediate-level radioactive waste (L/ILW) (NEPA, 1992), and the site selection, construction and operation of the disposal facilities under the liability of the China National Nuclear Corporation (CNNC), and the activities of regulatory control under the liability of the state Environmental Protection Administration (SEPA). In 2002, the law on Radioactive Pollution Prevention became effective in China. This is a basic law on radioactive management. The criteria related to Low-and Intermediate-level radioactive waste disposal, such as waste acceptance and package criteria, have been issued. The standards related to Low and intermediate level waste disposal in China are listed as following:

- GB 11928-89, "Regulations for interim storage of Low-and Intermediate-Level Radioactive Solid waste".
- --- GB 12711-91. "Standard of safety for Low and Intermediate-Level solid radioactive waste packages".
- GB/T 15950-1995. "General requirements for environmental radiation monitoring around near surface disposal site of Low-Intermediate level radioactive solid waste".
- GB 13600-92, "Regulations for Disposal of Solid Low-and Intermediate Level Radioactive wastes in rock cavities".
- GB 14569.1~14569.2-93. "Characteristic requirements for solidified waste of Low and Intermediate Level Radioactive waste-Cement solidified waste and plastic solidified waste".
- --- GB 14569.3-1995. "Characteristic requirements for solidified waste of Low and Intermediate Level Radioactive waste-Bitumen solidified waste".
- GB 16933-1997. "Acceptance criteria for near surface disposal of Radioactive waste".
- EJ 914 -2000. "Concrete container for Low-and Intermediate-Level Radioactive solid wastes".
- 3. WORK CARRIED OUT
- (1) Information survey on methodology and technology related to LILW disposal and on the topic of on-site disposal practices,
- (2) Field work for ER on-site disposal facility,
- (3) Typical radioactive survey on on-site disposal facility,
- (4) Environmental impact assessment on the option of on-site disposal,
- (5) Verification inspection of effectiveness of the disposal facility.
- (6) Optimization of design of engineered structures with respect to the on-site disposal of decommissioning radioactive waste.

# 4. THE OPTION OF ON-SITE DISPOSAL OF DECOMMISSIONING WASTES AND RELATED PRACTICE IN CHINA

On-site disposal is not without precedence because there have been instances, particularly in the 1960s and early 1970s, where it has been performed successfully, and without undue public concern. Activities ranged from in situ disposal of entire facilities or portions thereof to disposal within the site boundary of major components such as the reactor pressure vessel or steam generators [1-5]. These activities have largely been restricted to small facilities (e.g. research reactors or demonstration plants). Moreover, a number of studies and proposals have explored variations within the strategy [6].

Until now, only a few decommissioning activities of nuclear facilities have been completed in China, although most facilities have been earmarked for decommissioning in the near future. This paper will focus on reviewing on-site disposal of decommissioning waste in the world to understand the advantages and limitations of the option of on-site disposal of decommission waste, and study the feasibility of the option of on-site disposal of decommission waste in China. The related results might be useful for future decommission of nuclear facilities.

The ER facility is a nuclear fabrication facility in China, which is located at northwest China. The on-site disposal facility was constructed to disposal low-level operational wastes and contaminated soil and demolition debris during decommissioning process. The primary contaminants of decommissioning wastes are depleted uranium.

# 4.1. Site background

Siting was conducted according to the national standard GB9132-88, "Regulations for Shallow Ground Disposal of Solid Low-and Intermediate-Level Radioactive waste". The site is located in northwest China. Mild hypsography and low hills are dominant landscapes in this area. The climate here is characterized of the highland continental climate without four clearly defined seasons. Mean annual temperature is  $-1.7^{\circ}$ C. Annual precipitation is about 600 mm with most in summer. The site has strong solar radiation and mean annual global solar radiation is up to 6000 ~ 7000 MJ m<sup>-2</sup>.Vegetation in this area is typical frigid vegetation, such as Potentilla fruticisa shrub, Kobresia humilis meadow and swamp meadow. The main soil type is alpine meadow soil, alpine scrubby meadow soil, and swamp soil.

The upper crust of the site is composed of early Proterozoic crystalline metamorphic rock, mid-late Proterozoic sedimentary formation, Palaeogene sedimentogeneous rock, and Quaternary weathering rock. The disposal pit is constructed within Palaeogene sedimentogeneous rock. From surface to bottom, the stratums are topsoil, silt soil, yellow earth (Quaternary weathering rock, the thickness is from 1.5m to 3.5m) and red gravel, coarse sandstone, siltstone, and mudstone (Palaeogene sedimentogeneous rock). The hydraulic conductivity of the surrounding formation is from  $10^{-5} \sim 10^{-8}$ m/s.

The water table is in range of 10m to 15m below the surface. The base of disposal pit is about 5m higher than the local water table and the site is far away from the surface water (>650m). The groundwater discharges mainly through surface flow or runoff. The pH and total dissolved solids of the groundwater is  $7\sim7.9$  and  $0.09\sim0.27$ g/l respectively. The analytic results of different water samples at the Site are listed at Table 1.

No.	Types	pН	Total dissolved solids	Chemical component (Unit: mg/L)					
				HCO <sub>3</sub> -	Cl	$SO_4^{2-}$	Ca <sup>2+</sup>	$Mg^{2+}$	$K^++Na^+$
1	Spring	7.7	0.27	285	10.2	16.3	80	11.9	5.2
2	Spring	7.0	0.27	282	11.9	12.1	80.2	13.8	6.5
3	Spring	7.9	0.27	233	11.6	9.0	60.4	11.8	8.0
4	Well water	7.8	0.16	160.6	12.5	10.3	47.5	7.3	5.2
5	River water	6.9	0.22	200.3	18.0	14.6	56.5	13.8	7.5
6	Snow water	7.4	0.09	54.79	2.91	5.93	17.82	1.2	2.65

Table 1. Analytic results of water samples at the site

The site is suitable for the construction of a disposal pit, reasons are listed as following:

- (1) High stability in regional geological structure, no active faults;
- (2) Low population density;
- (3) Weak seismic activity;
- (4) No poor engineering geological conditions such as landslide, avalanche, liquefy of sandy soil, etc.);
- (5) The water level is deep, thick clay layer with low permeability in the vadose, strong retardation of nuclides from migration,
- (6) Far away from surface water, no flood risk.

# 4.2. Inventory of wastes

During the decommissioning stage, more than 25 000 pieces of equipment, 12 000-meter pipeline system in length and 180 000m<sup>2</sup>-contaminated area were decontaminated by mechanical and chemical methods. The contaminated metal will be recycled and the wastes with high radium content (>7Bq/g) were packaged and transported to low-intermediate-level radioactive wastes repository for disposal. The wastes with low radium content (>7Bq/g) were disposed of in the on-site disposal facility. According to the national standard GB 9133, "The Standard for Classification of Radioactive Waste", the wastes to be disposed of include: Total 6100 cubic meter, depleted uranium wastes occupy 94.2% (specific activity (SA) ~3.15×10<sup>3</sup>Bq/kg), the others are radium wastes (~5.7%) and cerium wastes (~0.05%). The types of the wastes include some operational wastes during the history of ER facility and large quantity of very low level decommissioning radioactive wastes, i.e. sand, soil, some brick, and bitumen materials. Disposal volume is about 60 000 m<sup>3</sup>.

### 4.3. Engineered structure

The disposal pit was constructed by excavation. The depth of the disposal pit is about 7 meters. Two well known release scenarios should be envisaged: groundwater releases, and loss of containment or cover by external events. In order to render the nuclides to escape from disposal pit into groundwater, the bottom of the disposal pit was filled with 50cm compacted local red soils (The hydraulic conductivity is less than  $10^{-10}$  m/s) and the side walls are filled with 30cm compacted clay (dry density is 1.59 g/cm<sup>3</sup>).

The drainage ditches are constructed to discharge rainwater and the surface water. The slope of the ground surface in the disposal zone is  $3^{\circ} \sim 5^{\circ}$  in length and  $10^{\circ} \sim 15^{\circ}$  in breadth, which favours the rainfall flowing to drainage ditches other than advancing by infiltration.

The wastes are disposed layer by layer. The thickness of each layer about 20cm and there are 25 layers in total. After the first waste layer was disposed, water was sprayed into the waste and compaction was conducted; this was repeated for the next layer.

In order to reduce potential erosion, bio-intrusion, and seepage, a multiple layer capping is designed, which is composed of 8 sub-layers, including (from bottom to surface) clay sub-layer (45cm), 30% limestone + 70% soil (15cm), geo-textile, 30% limestone + 70% soil (15cm), cobble (thickness: 50cm;  $\emptyset$ : 5~10cm), gravel (thickness: 30cm;  $\emptyset$ : 1~5cm), clay sub-layer (50cm), and topsoil for vegetation (230~370cm). The capping was constructed layer by layer.

# 5. ENVIRONMENTAL IMPACT ASSESSMENT

The main content of the EIA for the on-site disposal of decommissioned wastes includes (1) Description of the project; (2) Legal framework; (3) Natural & social environment of the decommissioning facility; (4) Initial radiological state; (5) Description of the decommissioning facility; (6) Selected decommissioning strategy and its alternatives; (7) EIA of the decommissioning process; (8) Radiological state after decommission; (9) Post-closure environmental monitoring plan; (9) Quality assurance; (10) Conclusions & recommendations; (11) References.

EIA shows that the maximum individual effective dose is 2.2 mSv during decommissioning process, which is lower that of national regulation (related limitation is 5 mSv). The collective effective dose is  $7.4 \times 10^{-2}$  person Sivert, which is also lower than limitation of national standard [7]. After decommission, the maximum individual effective dose is 0.02 mSv for the residents nearby, which is lower that of national regulation (related limitation is 0.25 mSv). The collective effective dose is  $1.6 \times 10^{-2}$  person Sivert, which is also lower than limitation of national standard [7].

# 6. VERIFICATION INSPECTION

The objective of inspection and verification is to ensure confidence in the effectiveness of the on-site disposal facility. Inspection and verification conducted at the site includes the following environmental monitoring activities, e.g. liquid pathway (groundwater and surface water), air pathway (radiological air particulates, radon, and direct radiation on the surrounding public and environment), sediment etc. The measurements include external irradiation measurement, water, soil, grass, meat, rice & vegetable sample collection. Since the depleted uranium is the key nuclide for the site. The uranium contents of sample collected in the surrounding environment are shown in Table 2. No abnormal results were found during external irradiation measurement, and the engineering structure is in good condition.

The inspection and verification shows that the engineering measures can successfully assure the isolation of decommissioning wastes from the surrounding environment. The gamma survey shows that the penetration radiation rate on the surface of the cover is just the similar to that in the surrounding environment. Samples taken and analyzed from the surrounding environment has not shown any migration of the uranium from the disposal pit to the environment, proving the effectiveness of the on-site disposal facility.

Sample	Before Decommission	After Decommission		
Water (µg/L)	2.8~4.4	2.2~4.4		
Water in the lake(µg/L)	15.4~17.5	14.5-18.7		
Soil (Bq/kg)	20.2~59.0	21.9~58.7		
Pasturage (Dry µg/kg)	2.4~10.4	3.4~9.8		
Beef (Wet µg/kg)	1.5~4.7	2.3~3.5		
Mutton (Wet µg/kg)	2.2~4.1	2.2~3.1		
Fish (Wet µg/kg)	3.8~5.2	3.6~5.4		
Rice (Wet µg/kg)	1.2~4.0	1.5~2.2		
Vegetable (Wet µg/kg)	0.5~1.8	0.5~1.6		
Milk (Wet µg/kg)	2.6~3.6	2.1~3.9		

Table 2. Uranium content samples collected in the surrounding environment

# 7. CONCLUSION

On-site disposal, which implies that most of the radioactive material will be permanently emplaced in an engineered structure near the surface, may sometimes be a reasonable option for the specific decommission waste that is characterized by large volume and very low-level nuclear waste. At the ER site, the on-site disposal facility was constructed to dispose lowlevel operational wastes and contaminated soil and demolition debris during decommissioning process. The disposal pit was constructed by excavation. The depth of the disposal pit is about 7 meters. In order to retard the nuclides from escaping from the disposal pit into groundwater, the bottom of the disposal pit was filled with 50cm compacted local red soils and the side walls were filled with 30cm compacted clay. The drainage ditches are constructed to discharge rainwater and the surface water. The slope of the ground surface in the disposal zone is  $3^{\circ} \sim 5^{\circ}$  in length and  $10^{\circ} \sim 15^{\circ}$  in breadth, which favours the rainfall flowing to drainage ditches rather than advancing by infiltration. The wastes are disposed layer by layer. The thickness of each layer about 20cm and there are 25 layers in total. In order to reduce potential erosion, bio-intrusion, and seepage, a multiple layer capping is designed, which is composed of 8 sub-layers. Inspection and verification shows that the engineering measures can successfully assure the isolation of decommissioning wastes from the surrounding environment. The gamma survey shows that the penetration radiation rate on the surface of the cover is just the similar to that in the surrounding environment. A set of analysis of samples from the surrounding environment hasn't shown the migration of the uranium from the disposal pit to the environment, which proves that effectiveness of the on-site disposal facility.

With regard to the ER on-site disposal facility, advantages & disadvantages of on-site disposal are summarized in Table 3 with refer to the IAEA technical report [6].

ADVANTAGES	DISADVANTAGES
Reduced cost	Need for long term maintenance and surveillance
<ul> <li>Minimal off-site transport and disposal</li> </ul>	
Reduced worker dose	A little difficulty in licensing and gaining public
<ul> <li>Reduced waste handling</li> </ul>	acceptance
Reduced public interaction	Very long term site commitment
<ul> <li>Fewer off-site activities</li> </ul>	
Early on-site disposal may reduce	Site size may be too small
monitoring costs	
<ul> <li>Less surveillance than safe enclosure</li> </ul>	
Early release of parts of site for non	May only be acceptable for certain nuclides
nuclear use	
<ul> <li>Reduces boundary of licensed site</li> </ul>	

# Table 3. Summary of advantages & disadvantages of on-site disposal

### ACKNOWLEDGEMENTS

The authors are grateful to Mr. Wang & Mr. Fan who provided technical support. Thanks are also given to Messrs. Jiao, Zhong, Liang and. Guo who provided support for fieldwork. The International Atomic Energy Agency and CNNC are thanked for financially supporting the research.

#### REFERENCES

- [1] PUERTO RICO WATER RESOURCES AUTHORITY, UNITED NUCLEAR CORPORATION, Boiling Nuclear Super heater Power Station Decommissioning Final Report, Docket 1154-2, WRA-B-70-500 (1970).
- [2] WHEELOCK, C.W., HEWSON, R.A., Retirement of the Piqua Nuclear Power Facility, (Proceedings American Nuclear Society Conf. on Reactor Operating Experience, San Juan, 1969), American Nuclear Society (1969).
- [3] WHEELOCK, C.W., Retirement of the Piqua Nuclear Power Facility, AI-AEC-12832, Atomics International, Canoga Park, CA (1970).
- [4] ATOMICS INTERNATIONAL, Retirement of Hallam Nuclear Power Facility, AIAEC-12709, Atomics International, Canoga Park (1970).
- [5] MESERVEY, R.H., OMRE and Hallam Decommissioning Projects at the Idaho National Engineering Laboratory (Proc. American Nuclear Society topical meeting, Sun Valley, ID, 1979), Plenum Press, NY (1980).
- [6] INTERNATIONAL ATOMIC ENERGY AGENCY, On-Site Disposal as a Decommissioning Strategy, IAEA-TECDOC-1124, IAEA, Vienna (1999).
- [7] GB 18871-2002. Basic standard for protection against ionizing radiation and for the safety of radiation sources. General Administration of Quality Supervision, Inspection and Quarantine of the People's Republic of China. 2002. (In Chinese)
- [8] NEPA, China National Environmental Protection Agency, "Environmental Policy of China on Low-and Intermediate-Level Radwaste Disposal", June 24, 1992. (In Chinese).
- [9] Wang Xiande. 1997. Low –and intermediate-level radwaste disposal in China. http://www.wmsym.org/wm97proceedings/sess13/13-35.htm.
## GERMANY

#### MANAGEMENT OF DECOMMISSIONING WASTE IN GERMANY

**F. Borrmann**, Federal Office for Radiation Protection, Department for Nuclear Safety, Section Decommissioning of Nuclear Installations, Salzgitter

**P. Brennecke**, **W. Koch, K. Kugel, S. Steyer**, Federal Office for Radiation Protection, Department for Safety and Nuclear Waste Management, Division for Safety and Planning, Salzgitter

## Abstract

Over the past two decades, Germany has gained a substantial amount of experience in the decommissioning of nuclear facilities of different types and sizes. Many research reactors and all prototype nuclear power plants, as well as a few larger nuclear power plants and fuel cycle facilities, are currently at varying stages of decommissioning. Several facilities have been fully dismantled and the sites have been cleared for reuse. The decommissioning projects comprise 18 power and prototype reactors, 33 research reactors and 11 fuel cycle facilities which are being or have been decommissioned [1]. In the future, further nuclear power plants will be shut down and decommissioned in accordance with Germany's energy policy to phase out the use of nuclear power for commercial electricity generation as given in the April 2002 amendment of the Atomic Energy Act [2].

Radioactive waste, from operations as well as from decommissioning activities, is to be conditioned in such a way as to comply with the waste acceptance requirements of a repository. In Germany, all types of radioactive waste (i.e., short-lived and long-lived) are to be disposed of in deep geological formations. A distinction is being made for heat generating waste (i.e., high level waste) and waste with negligible heat generation (i.e., low level and intermediate level waste). Radioactive decommissioning waste is waste with negligible heat generation.

Waste acceptance requirements of a repository are of particular importance for the conditioning of radioactive waste, including decommissioning waste. The waste acceptance requirements, as they resulted from the Konrad licensing procedure, are being applied by the waste generators for the conditioning of decommissioning waste. Compliance with these requirements must be demonstrated through the waste package quality control, even if the waste will be disposed of in the future.

In 2002 the Konrad repository was licensed for the disposal of all types of waste with negligible heat generation. Konrad is an abandoned iron-ore mine to be reconstructed for use as disposal facility. It is not yet in operation as the license is actually examined by court. Dismissal of legal action is an important prerequisite for the realization of the Konrad project. Furthermore, the Federal Government needs to take a final decision on the reconstruction and operation of the Konrad repository.

## 1. SCOPE OF THE PROJECT

The scope and objectives of the project are to provide an overview over the German experience in the decommissioning of nuclear facilities and the management of the resulting radioactive materials, in particular the conditioning and disposal of radioactive waste and the release of materials, buildings and sites from nuclear regulatory control.

The report provides information on legal, regulatory and technological aspects of the decommissioning of nuclear facilities as well as strategy considerations. It provides information about some examples of decommissioning projects in various stages, but does not give details for specific issues. The objective of these examples is to provide information about the amount and types of waste that are to be expected from decommissioning of the respective nuclear facility type. The report gives some hints on the amount of waste that will be suitable for clearance from regulatory control.

The report also provides information on the disposal of radioactive waste, based on the experience gained and knowledge resulting from the licensing of the Konrad repository, in particular on the waste acceptance requirements and the waste package quality control.

All these facets are presented from the regulators point of view, not on a technical scale. The projects have been selected to represent a broad range of nuclear installations. Nevertheless, not all desirable information was available for publication without restrictions. Some of the information (e.g., cost) is subject to nondisclosure from the regulators point of view. The information might, however, be provided on request by the operators themselves, depending on the individual information policy of the operator.

# 2. KEY FINDINGS

# 2.1. Strategy selection and funding

One of the key parts of decommissioning is the selection of a decommissioning strategy. From the three available strategies – immediate dismantling, safe enclosure and entombment – the latter is not accepted as an option in Germany as it implies installing a near surface repository. However, the regulator does not interfere in the operator's decision between intermediate and deferred dismantling, as long as the requirements for nuclear, worker's and public safety are fulfilled. This is for example the case if safe enclosure would be selected for a fuel cycle facility that had manufactured plutonium containing fuels. Due to the dose built-up by daughter nuclides, the decommissioning process would be gravely complicated and higher doses for the workers would result from a deferral.

The selection between immediate dismantling and safe enclosure (deferred dismantling) has to be considered under the following aspects:

- --- Decay of radionuclides / Radiation protection / Technology
- Waste amounts
- Cost
- Funding
- Radiological characterization
- Use of existing components
- Qualified staff
- Existence of an operator
- Termination of the nuclear liability
- Social impact
- Public acceptance
- Waste disposal facility
- Waste acceptance requirements

In most cases, immediate dismantling has been chosen in Germany even despite the fact that an interim storage facility is needed to bridge the time until a repository will be available in Germany.

The power utilities manage decommissioning and dismantling (with the exception of the disposal of radioactive waste) at their own responsibility, being supervised by the competent authorities. The cost is covered by provisions built up during the operating period of the respective plant. The allocation of reserves for the decommissioning of nuclear power plants covers all cost associated with dismantling of the plant itself. This includes the cost of the post-operational phase in which the facility is prepared for dismantling after its final shutdown (including removal of fuel assemblies and operational wastes), the cost for the licensing procedure and supervision, the cost of dismantling (dismantling and interim storage of all components and all buildings of the controlled area), and the cost of the interim storage and disposal of all radioactive wastes from decommissioning. The total amount of cost is estimated from cost studies which are updated regularly by the utilities, considering technical advancements and general price trends. These cost estimates are checked by the fiscal authorities.

The decommissioning of publicly-owned facilities is financed from the current budget. For most projects the Federal Government covers the bulk of the cost. Financing includes all expenses incurred for the post-operational and transition phase, disposal of the spent fuel elements, execution of the licensing procedure, dismantling of the radioactive part of the facility, and disposal of the radioactive wastes, including all preparatory steps.

## 2.2. Technical issues

The technical issues of decommissioning –decontamination and dismantling techniques – are solved to a large extent. Of course, there is still some potential for optimization regarding cost, timescale, waste volume and radiological protection.

A broad range of cutting technologies have been developed or adapted for decommissioning purposes (e.g. dry wire and plasma cutting). Special emphasis must be laid on techniques with low production of aerosols, such as nibblers for metal cutting or on techniques that can be adapted for underwater use. The example of the WAK reprocessing pilot plant shows that clear limits for the use of semi-remote controlled or full remote controlled techniques and structured planning can reduce the dose for workers significantly.

Decontamination technologies have been adapted and are widely automated (e.g., automated shaving devices for walls, floor and ceiling).

Further technological efforts have been undertaken in the field of clearance and data management. Buildings and ambient areas are usually measured by collimated gamma-spectroscopy. Clearance measurement devices with an adequate shielding and multiple detectors are used by default for the release of bulk material. Newer applications use individual nuclide vectors for small sectors of the respective plant, which are gained by measurement and sampling in advance to the decommissioning process. These nuclide vectors, gathered in a data processing system, allow the release measurements to distinguish much sharper for each fraction whether release is possible or not.

## 2.3. Waste amounts and pathways

For the estimation of waste amounts from decommissioning, a study for two different types of reference reactors, one pressurized water reactor (PWR) and one boiling water reactor (BWR), has been carried out and is regularly updated. It provides the volumes of conditioned waste, derived from the masses of the main components of the plants. There are about 5 200 m<sup>3</sup> of radioactive waste arising from the decommissioning of a commercial 1 200 MW PWR in case of a direct dismantling strategy. A deferred dismantling strategy (30 years of safe enclosure) will reduce the amount of radioactive waste to about 4 300 m<sup>3</sup>. This represents about 2-3 % of an anticipated mass of the controlled area of about 150 000 Mg. In case of the BWR, there are 6 800 m<sup>3</sup> of radioactive waste awaited in case of immediate dismantling, and 5 400 m<sup>3</sup> from a deferred dismantling (30 years of safe enclosure). This also reflects about 2-3% of the anticipated total mass of the controlled area (about 230 000 Mg).

This may further vary according to the type of facility. The WWER reactor types in Greifswald (KGR) and Rheinsberg (KKR) have much higher masses (KGR 1-6 comprise a total mass of t about 1 800 000 Mg). Approximately 1% of this mass (18 000 Mg) will be radioactive waste. This would be some 3 000 Mg per 440 MW block. On the other end of the scale, the decommissioning of a SUR-100 research reactor, comprising a total mass of 10 Mg, did not result in any radioactive waste at all, beside the fuel plates and the neutron source.

Most of the primary waste is subjected to a clearance procedure and can be released from regulatory control. In Germany, there are several clearance pathways: unrestricted clearance and three restricted clearance pathways – metals for melting, solids and liquid for conventional disposal and buildings for demolition. In these cases, no conditioning of the wastes is allowed. The following graph (Figure 1) shows the clearance process in Germany.



FIG. 1. Clearance of decommissioning waste in Germany.

For NPP decommissioning, the most important pathways are clearance for melting and unrestricted clearance. For nuclear fuel cycle facilities, the restricted clearance for disposal plays an important role, because of widespread low-level contamination with long-lived nuclides. A substantial fraction of this waste is disposed of in conventional subsurface repositories.

# 2.4. Interim storage

As a repository is currently not available in Germany, decommissioning waste has to be stored for the time being. The interim storage facility will remain on site after the decommissioning is completed. There are several options to solve the storage problem. One solution is to use existing buildings as it has been done for the Wuergassen NPP. Here, the RTS/RHR-building (the reactor trip system and the residual heat removing system building) and the existing low level waste storage building are used. The license for the storage facility is then usually covered by the decommissioning license. Nevertheless, the storage facility has to meet the requirements valid at the time, when the decommissioning license is granted. This may cause the need for a general refurbishment of existing buildings. Additional backfitting and conversion may be needed if the former purpose of the building was not storage. A major advantage is that at the end of the storage period, the respective waste amounts as well as the dismantling, decontamination and clearance measurement efforts may be lower, as the existing building would have been subject to decommissioning anyway. Storage facilities from operation may also be sufficient to store the waste from decommissioning. This might be the case for smaller research reactors.

Another solution is to use external or centralised storage facilities. This is (due to the waste transport) mainly applicable for research reactors and especially for research centres. The Karlsruhe Research Centre has shown that the decommissioning waste from several facilities (research and prototype reactors as well as the reprocessing pilot plant) can be conditioned and stored in a centralised facility.

The third solution is the construction of a new storage facility as it will be done for the decommissioning waste from the Stade NPP. In this case, the main reason is to gain space, as the plant itself is rather narrow. The license for the storage facility can be covered by the decommission license or may be regulated in an independent license. The latter has the advantage, that the regulatory burden on the site might be smaller after the deregulation of the plant itself.

# 2.5. Conditioning and waste acceptance requirements

Presently in Germany, neither obligatory waste acceptance requirements nor an operational repository are available. This will inevitably affect the conditioning of radioactive waste. On the one hand, the non-availability of a repository means that the aspect of long term storage has to be taken into account in waste conditioning. On the other hand this situation will necessitate the current waste conditioning measures basically being planned and carried out in such a way that, depending on necessities and licensing-relevant framework, certain flexibility should be granted with respect to future waste conditioning steps. This should be seen as a contribution to reduce both the effort to meet future waste acceptance requirements and the additional radiation exposure of the personnel.

Only solid or solidified radioactive waste will be accepted for disposal in deep geological formations; liquid and gaseous waste is excluded from acceptance. The controlled safe disposal of radioactive waste, therefore, requires its conditioning prior to disposal.

The results of the Konrad site-specific safety assessment have been converted into both a design of the surface and underground facilities as well as in a system of waste acceptance requirements. This system is formulated is such a way that it first describes the general disposal-related aspects and the general requirements to be fulfilled by the waste packages, and then develops into more specific requirements on the waste forms, the waste container/packaging, the activity limitations on individual radionuclides, the documentation and the delivery of waste packages. A survey of the organization of the Konrad waste acceptance requirements is given in the following:

- (1) General basic requirements on radioactive waste to be disposed of
- (2) General requirements on waste packages
  - Local dose rate
  - Surface contamination
  - Depressurized delivery
- (3) Requirements on waste forms
  - Basic requirements
  - Waste form groups
  - Exhausting of activity limiting values
  - Filling of waste packages
- (4) Requirements on waste containers/packagings
  - Basic requirements
  - Waste container classes
  - Incident resistant packagings
  - Inner containers
- (5) Activity limitations
  - Permissible activities for individual radionuclides per waste package
  - Total activities
  - Declaration of radionuclides
- (6) Delivery of waste packages
  - Compliance with transport regulations
  - Permits
  - Marking of waste packages
  - Requirements on shipping units

## 2.6. Waste packaging quality control

Generally speaking, the BfS regulations on waste package quality control of radioactive waste with negligible heat generation admit two methods of proving that the waste acceptance requirements are met:

- --- Random sample testing of waste packages already produced, or
- Qualification of conditioning techniques and determination of accompanying control measures to be carried out.

Both alternatives were examined in detail and confirmed by the Lower Saxonian Ministry for Environment as competent licensing authority for the Konrad repository.

According to Section 74, paragraph 2 of the Radiation Protection Ordinance (StrSchV [4], methods that have been approved by the Federal Office for Radiation Protection have to be applied for the treatment and packaging of radioactive waste to produce waste packages that are suitable for disposal. According to the "Guideline on the Control of Radioactive Waste with Negligible Heat Generation that is Not Delivered to a State Collecting Facility (Waste Control Guideline) [5]", qualified techniques are to be applied where possible for pre-treatment and conditioning.

The application of specific waste package quality control measures prior to emplacement of the waste packages in a repository has proven successful in practice during emplacement operations in the Morsleben repository for radioactive waste. Co-operation between all the institutions involved has likewise worked well. The experience thereby acquired does not suggest any diverging from these techniques.

# 2.7. Waste disposal

Between 1994 and 1998 waste from decommissioning of NPPs was disposed of in the Morsleben repository (ERAM). This waste originated from the following installations:

- NPP Gundremmingen, unit A (KRB-A, 250 MW)
- NPP Niederaichbach (KKN, 106 MW)
- Hot Steam Reactor Großwelzheim (HDR, 25 MW)

The KKN and HDR were completely decommissioned. Decommissioning of KRB-A has been started 1983 and is not finished yet.

A total of 1654 waste packages containing LLW and MLW decommissioning waste were disposed of in the ERAM. Most of these packages were in 200-l-drums, with 359 packages in 400-l-drums. The main components were rubble from dismantling of the biological shield of all three NPPs as well as compacted mixed waste from the NPP Gundremmingen.

The decommissioning waste originating from these installations was conditioned according to the ERAM waste acceptance requirements. This waste has been disposed of in the ERAM without any problem.

# 3. CONCLUSIONS

# 3.1. Regulatory framework

The German regulatory framework, despite its rather complicated structure, proved to adequately regulate the decommissioning of a broad variety of nuclear installations.

# **3.2.** Strategy selection

Of the three decommissioning strategies, entombment is not accepted in Germany. The regulator does not usually interfere with the decision of the applicant with the selection of a strategy, as long as safety is guaranteed and no difficulties are presented by the decommissioning project as a whole.

Immediate dismantling is the widely applied strategy in Germany. Only 3 research reactors and two power reactors are in safe enclosure at this time.

Nevertheless, there has been a tendency in the last few years — at least for publicly financed projects — to remove large components (in particular reactor pressure vessels - RPV) as a whole and store them in an external storage facility. A leading role is played by EWN (Energiewerke Nord GmbH, a federal decommissioning operator), who has applied for the removal of the reactor pressure vessels for the Greifswald, Rheinsberg and Jülich reactors. In the case of Rheinsberg, the whole RPV will be transported to the interim storage facility in Lubmin by railway. The advantage of a deferred dismantling — dose reduction of the highly activated parts by decay — is combined with an immediate dismantling of the other components and buildings. This could lead to a shorter timescale, but higher waste volumes, than for deferred dismantling. This strategy is only applicable because an adequate storage facility is already available (Zwischenlager Nord).

# 3.3. Waste storage

The lack of a repository leads to the interim storage of decommissioning wastes for a time span of several tens of years. As has been shown, there are several possibilities to solve this storage problem: on-site or off-site storage, either in existing, backfitted or newly built facilities. Nevertheless, interim storage needs to be addressed by adequate criteria for the conditioning (i.e., processing and packing) of wastes.

## **3.4.** Waste conditioning

Presently, radioactive waste in Germany is mainly conditioned according to the Konrad waste acceptance requirements. Nevertheless, there is no guarantee that the waste will not have to be reconditioned. Therefore, it is important that no steps are performed anticipating or requiring a final conditioning, unless necessary.

## 3.5. Waste clearance

The tables provided in Appendix III of the Radiation Protection Ordinance (StrlSchV) comprise a variety of values for clearance of materials from decommissioning. This was an important step, as the decision, which values were adequate had to be taken case by case before. The new regulation allows the decision immediately if the values of appendix III StrlSchV are fulfilled. Otherwise, it also allows a case by case decision, if it can be proven by the applicant that the  $10\mu$ Sv - concept will be met by higher values.

## 3.6. Final disposal

No problems occurred during the disposal of decommissioning waste in the Morsleben repository (which is no longer operational). The decommissioning waste resulted from three different NPPs and consisted mainly of concrete rubble and mixed waste.

A repository for waste with negligible heat generation currently does not exist in Germany. The former iron ore mine Konrad was licensed, but the decision is still subject to court examination.

## REFERENCES

Useful links:

- [1] Brochure Decommissioning of nuclear facilities (English): http://www.bmu.de/files/pdfs/allgemein/application/pdf/boschuere\_stilllegung\_en.pdf.
- [2] Handbook on Nuclear Safety and Radiation Protection, 31. Supplement (06/05) (partially English): <u>http://www.bfs.de/bfs/recht/rsh</u>.
- [3] Homepage of BfS (Department for Nuclear Safety) (partially English): <u>http://www.bfs.de/kerntechnik?setlang=en</u>.
- [4] Homepage of BfS (Department for Safety and Nuclear Waste Management) (partially English): <u>http://www.bfs.de/endlager?setlang=en</u>.
- [5] Homepage of BfS (Section Decommissioning of Nuclear Installations) (partially English): <u>http://www.bfs.de/kerntechnik/stilllegung?setlang=en</u>.
- [6] Report under the Convention on Nuclear Safety by the Government of the Federal Republic of Germany (English): http://www.bmu.de/files/pdfs/allgemein/application/pdf/nuclear\_safety.pdf.
- [7] Report under the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management by the Government of the Federal Republic of Germany (English):
   http://www.bmu.de/english/nuclear\_safety/application/pdf/2nationaler\_bericht\_atomene

http://www.bmu.de/english/nuclear\_safety/application/pdf/2nationaler\_bericht\_atomene rgie\_en.pdf.

## HUNGARY

## INVESTIGATIONS OF THE DISPOSAL ASPECTS FOR LOW AND INTERMEDIATE LEVEL WASTES ORIGINATING FROM THE DECOMMISSIONING OF PAKS NPP

#### S. Patai Szabó, F. Takáts,

TS Enercon Kft., Budapest

### Abstract

Hungary so far has only very limited experience in the decommissioning of nuclear facilities, gained from decommissioning a few small nuclear facilities. It is planned to decommission the four operational power reactors, two research reactors, an Independent Spent Fuel Store and other facilities in the future.

The Nuclear Safety Directorate of the Hungarian Atomic Energy Authority (HAEA NSD) requires that a preliminary Decommissioning Plan and a valid strategy of decommissioning exist for all nuclear facilities in the country. The Decommissioning Plans (available presently only for the NPP and the Independent Spent Fuel Store) provide an estimated inventory and source term for the decommissioning wastes.

Radioactive waste, from operations as well as from decommissioning activities, is to be conditioned in such a way as to comply with the waste acceptance requirements of the existing or future repository. L/ILW radioactive waste from the operation and decommissioning of the Paks NPP are to be disposed of in a subsurface disposal facility, while HLW will be placed in a repository to be located in deep geological formation.

The general scope of the Project was to further collect data about the types and amounts of wastes generated during decommissioning, to improve the calculation of radionuclide inventory, to assess performance of waste packages (corrosion and gas generation).

#### 1. SCOPE OF THE PROJECT

The scope of the research project was to:

- (1) Collect data on the inventory and characteristics of wastes arising during decommissioning of Paks NPP
- (2) Develop a data base for the waste types generated during decommissioning
- (3) Further the use of scaling techniques to calculate the difficult-to-measure nuclides, which have a safety significance during their life in the repository
- (4) Improve the data base of NPP components to be disposed of after decommissioning
- (5) Improve knowledge about the inventory and source term of wastes already in the repository
- (6) Assess corrosion of waste packages and repository components
- (7) Further develop methods to evaluate gas generation of waste packages and their effect on the repository
- (8) Investigate performance of waste packages in repository conditions

The report provides summary information on the above issues. Specific details were covered in the submitted Progress Reports.

## 2. KEY FINDINGS

## 2.1. Decommissioning wastes

## 2.1.1. Inventory and characteristics of decommissioning wastes

Only relatively small amounts of waste are anticipated to be produced during the early stages of decommissioning, e.g. from the removal of fuel and the flushing out of the reactor coolant circuits.

With regard to the waste generated by the dismantling of the reactor, in accordance with general practice, a storage phase (safe enclosure) is foreseen. This period may last several decades to allow for short lived radionuclides to decay significantly. Even so, much larger volumes of low and intermediate level waste will be produced during decommissioning than from operation of the plant. According to calculations, the total amount of decommissioning waste for the NPP is as indicated in Table 1.

## Table 1. Total amount of NPP decommissioning waste

Waste type	Amount, m <sup>3</sup>
Low- and Intermediate Level Wastes	20,000
High Level Waste, (excluding spent fuel)	410

A spent fuel Modular Vault Dry Storage (MVDS) at the site serves for the interim storage of spent fuel. The expected lifetime of this facility is 50 years. The amount of decommissioning wastes from this facility is in Table 2.

#### Table 2. Decommissioning wastes generated in Modular Vault Dry Storage

Waste type	Amount
Low- and Intermediate Level Wastes	200 m <sup>3</sup>
Spent Fuel	1300 tHM

A detailed inventory of materials has been prepared for all buildings and systems of the NPP with regard to decommissioning. Design and operational data are the basic source of information for the inventory. In addition, the following basic assumptions and conditions are made in order to specify the conditions of decommissioning:

- (1) Operation of the Paks NPP will be terminated after the designed life time expiration, after the period of normal power operation. It is assumed that during operation there will have been no accidental situations and no accidental situation causes termination of operation.
- (2) All four Paks NPP units are subjected to decommissioning and the order of operation termination will be the same as it was for commissioning the units.
- (3) The substantial part of radioactive wastes from the Paks NPP decommissioning will be treatable and the final form will be acceptable for disposal in RAW repository planned in ÜVEGHUTA (subsurface disposal facility). The technological procedures and technical facilities ordinarily used in the course of normal Paks NPP operation will be utilized at maximum extent for treatment and conditioning of the radioactive wastes.

- (4) The radioactive wastes not meeting the criteria for disposal in above mentioned underground disposal facility will be transported and disposed in a deep geological repository located approximately 120 km from the Paks NPP.
- (5) During evaluation of decontamination the volume of used media, and the volume of produced wastes are determined by present day knowledge. Here the decontamination factors and the necessary number of decontamination cycles are considered, as well.
- (6) Spent fuel is cooled in the pool for 3 years. The spent fuel management, after its removal from unit into ISFSF, is not the subject of these preliminary calculations.
- (7) Demolition of all buildings is considered up to the depth 1 m. below –1 m level, only radioactive contaminated building materials will be removed.

Waste volumes were calculated in detail for the analyzed decommissioning options.

Isotopic composition of contamination of internal surfaces is supposed to be: 95%  $^{60}$ Co and 5%  $^{137}$ Cs for solid waste.

The above data generalises the separately compiled data sheets for each building and system, where the composition of wastes is also taken into account, i.e. whether the waste are concrete, steel (SS, or carbon steel), Al, brick, wood, or other.

The estimated average contamination of building surfaces (floors and walls) after reactor shutdown in separate active buildings was also calculated, and its range is between 2 and  $50 \text{ Bq/cm}^2$ . The number of contaminated pipe and electrical components was also calculated.

A certain amount of material from the material inventory requires special, industrially used treatment, e.g. during demolishing work. The buildings used for toxic waste storage, together with the storage basins and connected buildings require special treatment, as well as the oil storage tanks and the pipeline system for oiled water.

The assumptions for decontamination took into account the following elements:

- full system decontamination of primary circuit as a whole
- Internal pre-dismantling decontamination will be carried out in chosen technological systems.

# 2.1.2. Decommissioning waste disposal

Currently, radioactive wastes are disposed of at the facility in Püspökszilágy, which is currently the only site for disposal of radioactive waste in Hungary. The near surface facility was commissioned in 1976, and wastes from all research, medical and industrial applications have been sent there for treatment and disposal. The facility is composed of concrete trenches (vaults) and shallow wells (6 m deep) for spent sealed radioactive sources (SRS). The repository received solid low-level radioactive wastes also from Paks NPP between 1983 and 1997.

For L/ILW coming from the operation and decommissioning of the nuclear power plant, however, a new facility must be built. The anticipated total volume of packaged operation and maintenance waste during the planned 30 years of NPP operation is almost 21 000 m3. Current estimates of the total activity of this waste is 1700 TBq – consisting of solid waste, evaporates and ion-exchange resins. The waste is packaged in drums, compacted or grouted according to type. Drums with a low activity concentration could be disposed of directly in

the repository (although the present Hungarian regulations require retrievable solutions) and in other cases the drums will first be placed in concrete overpacks.

Construction of a mined subsurface repository for L/ILW at a site near Bátaapáti, also known as Üveghuta, takes place at some 200-250 m below the surface (i.e. at 0-50 m above sea level). The exact location of the disposal area will be defined after additional geological investigations have been performed and experience has been gained during tunnel construction. The layout of the subsurface facility will be affected by the geological environment and by the amount of waste. Currently, a tunnel-type arrangement appears more favourable. Both the waste drums and the disposal containers will be emplaced in disposal tunnels, so that radioactive isotopes escaping from the waste packages will be absorbed after a time by the clay backfill material (which contains bentonite) either around the waste packages or inside the containers. Some 10-20m thick granite pillars will separate the 6- or 10m-wide disposal tunnels, ensuring the mechanical stability of the repository. Design of the layout and of the tunnel characteristics will be refined after further geological investigations. Based on existing information on the geological situation at the site of a rock mass of the type to be expected at Üveghuta, a "design as you go" approach is being followed, adapting the design of the repository to the geological situation, as revealed during excavation.

Work is proceeding at the site investigation, and construction of 2 access tunnels was begun in 2005. It is anticipated that the application for a permit will be made to construct the repository in about 2008. The repository is due to be operational around 2010.

In the current repository design, disposal galleries (cross-section 85  $m^2$ ) extending from a central access tunnel will be constructed. The waste packages will be stacked within the disposal galleries and these are to be backfilled with crushed granite from the excavation mixed with 10% of bentonite.

## 2.2. Preparation of the decommissioning waste database

A detailed listing of calculated decommissioning wastes and their activity has been reported, based on the data calculated in the latest version of the Paks Decommissioning Plan [1].

Premise-oriented data were recorded on specific data sheets and an ORACLE – based date base was developed specifically for this purpose. In order to ensure the data sheets were accurate and consistent, detailed guidance for each type of data sheet was provided to personnel collecting the data.

The amount of wastes anticipated to be generated has been calculated by building and contaminated plant structure (e.g. reactor hall, auxiliary buildings, stacks, waste management building, etc), and by the type of material to be decommissioned (e.g. stainless steel, carbon steel, coloured metals, etc.).

The structure of the database has been reviewed by an international team within the RER-3-003 Technical Co-operation activity of the IAEA. A new, revised version will be prepared in 2006, and data collection will start soon thereafter.

## 2.3. Inventory and source term of wastes in the Püspökszilágy repository

An updated total inventory for the Püspökszilágy facility has been produced based on previous analyses [2-4]. Assessment of some of the options for waste retrieval will require an understanding of key elements of the inventory on a compartment-by-compartment basis. The

analyses used to determine the inventory are reasonably advanced; thus, this inventory data will be used for the assessment work.

The current radionuclide inventory is 2.61 E+14 Bq.

# 2.4. Corrosion of waste packages

The possible modes and rates of corrosion of mild steel and stainless steel in the radioactive waste repository located at Püspökszilágy have been calculated.

Leaving aside the issue of localised corrosion, the evolution of the repository leading up to saturation, i.e. dry, wet and saturated phases, were reviewed as follows. Initially, the repository is dry and atmospheric corrosion occurs, then water enters the repository and the full range of corrosion processes occur. Finally, the repository becomes saturated and only anaerobic corrosion processes are important.

If there is sufficient information to suggest that the repository rapidly saturates with water (which seems unlikely for the repository at Püspökszilágy), then the worst-case anaerobic corrosion rate values could be applied, rather than the aerobic ones. This is justifiable on the grounds that anaerobic conditions prevail for the most of the lifetime of the repository, and the duration of the dry and wet phases may be too short to have a significant impact on container performance.

## 2.5. Gas formation in L/ILW waste packages

Gas composition measurements by mass spectrometry analysis have been carried out on samples taken from the headspace of ten 200 l stainless steel drums containing LL/ILW generated and temporarily stored at Paks NPP.

Four drums contained compacted solid waste, three drums were filled with grouted sludge and three drums contained solid waste without compaction. The compacted wastes consist of contaminated trash and scrap, protective clothes, gloves, towels, mainly plastics, textile, wood and paper. The non-compacted wastes consist of debris of building material, out-of-use tools, mainly metals. The grouted sludge comes from cleaning (steam generators, floor in labs and workshops, etc.) and does not contain used ion exchange resin or evaporator concentrate.

The drums were equipped with a special gas outlet system to make repeated sampling possible. [5] The parameters of the investigated 10 drums are given in Table. 3.

Gases with concentration higher than 1% are presented in bold type. The measured pressure values in the drums and the pressure testing of an empty drum also proved that these drums are not hermetically closed.

Drum	Type of	Max. dose	Date of	Main components of bulk	Pressure of
code	waste	rate (nGy/h)	closing	gas	bulk gas (bar)
1T	Compacted	1200	25/03/98	N <sub>2</sub> , O <sub>2</sub> , Ar, CO <sub>2</sub>	1.00
2T	Compacted	4200	23/03/98	H <sub>2</sub> , N <sub>2</sub> , O <sub>2</sub> , Ar, CO <sub>2</sub>	1.10
3T	Compacted	2800	24/03/98	$H_2, N_2, O_2, Ar, CO_2, CH_4$	1.04
4T	Compacted	2500	19/03/98	H <sub>2</sub> , N <sub>2</sub> , O <sub>2</sub> , Ar, CO <sub>2</sub>	1.02
1NT	Non compacted	1200	03/04/98	N <sub>2</sub> , O <sub>2</sub> , Ar, CO <sub>2</sub>	1.00
2NT	Non compacted	4500	29/04/98	H <sub>2</sub> , N <sub>2</sub> , O <sub>2</sub> , Ar, CO <sub>2</sub>	1.07
3NT	Non compacted	2000	24/04/98	N <sub>2</sub> , O <sub>2</sub> , Ar, CO <sub>2</sub>	1.00
1 <b>S</b>	grouted sludge	2000	10/05/99	CH <sub>4</sub> , N <sub>2</sub> , O <sub>2</sub> , Ar, CO <sub>2</sub>	1.26
2S	grouted sludge	1800	06/05/99	CH <sub>4</sub> , N <sub>2</sub> , O <sub>2</sub> , Ar, CO <sub>2</sub>	1.08
38	grouted sludge	2500	08/10/99	N <sub>2</sub> , O <sub>2</sub> , Ar, CO <sub>2</sub>	1.00

Table 3. Parameters of the investigated drums

The gas formation processes vary from one drum to the other. Quantitatively, it can be stated that in general during the storage period the carbon dioxide content increased and oxygen content decreased.

As can be seen, hydrogen production was detected mainly in drums containing compacted wastes. The maximal value measured was less than 10%, while oxygen was depleted in these drums.

In two drums the rate of the gas generation was extremely high. In these cases methane and carbon dioxide were generated in rather high amounts, and the oxygen was depleted.

Carbon dioxide generation is characteristic for all types of drums, while methane formation is typical for drums containing grouted sludge.

The stable isotope ratio measurements proved that the surplus of  $CO_2$  measured in almost all drums is of organic origin.

Significant variation over time of tritium in the individual drums was observed. The maximal measured value was approx. 80 Bq/litre, the typical concentration values were between 0.1 and 10 Bq/litre.

The highest radiocarbon activity concentration measured in the headspace gases was about 3000 Bq/litre. Typical 14C activity values were between 1 and 100 Bq/litre.

The presented results represent only the first stage of a long-term investigation. Continuous sampling of these drums can help us to understand gas-formation processes in different type of LL/ILW. Further studies will focus on calculation of gas generation rates using the measurement data.

## 2.6. Performance of waste packages in repository conditions

Earlier, as a check on conditions within the vaults, a non-cemented (A05) compartment and a cemented (A06) compartment in an A vault were opened to check on the condition of the wastes, disposed 20 years ago.

The vaults were found to be dry and the vaults, cap and wastes were found to be in good condition, with little apparent degradation of either concrete or waste packaging.

## 3. CONCLUSIONS

Deferred decommissioning of the 4 Units of Paks NPP, after 70 years of safe storage of the Nuclear Island, is the reference scenario used for the relevant cost calculations.

The Decommissioning Plans (available presently for the NPP and the Independent Spent Fuel Store) provide an estimated inventory and source term for the decommissioning wastes, and the chronology of waste generation. In order to support a better understanding of the types of waste, the scope of the Project was also to improve the calculation of radionuclide inventory, to assess performance of waste packages (corrosion and gas generation).

The assessments presented here may be periodically updated in future, based on new information from the on-going investigations.

#### REFERENCES

- [1] Preliminary decommissioning plan of the Paks NPP, Decom Slovakia Ltd., TS Enercon Kft., 2003.
- [2] Safety Analysis of the Püspökszilágy Radioactive Waste Treatment and Disposal Facility: An Assessment of Post-closure Safety, Püspökszilágy Final Report, European Commission, PHARE 990167, PH4.12/95(01)N2, (2001).
- [3] Safety Analysis of the Püspökszilágy Radioactive Waste Treatment and Disposal Facility, Püspökszilágy Final Report, ETV-ERŐTERV, (2001) (in Hungarian).
- [4] Safety Analysis of the Püspökszilágy RWTDF, TS Enercon Kft., (2002) (in Hungarian).
- [5] MOLNÁR, M, PALCSU, L, SVINGOR, É, SZÁNTÓ, Zs, FUTÓ, I, ORMAI, P, "Gas formation in drum waste packages of Paks NPP", Proceedings of International Conference, Nuclear Energy in Central Europe 2000, Bled, Slovenia, 11-14 Sept., (CD-ROM), (2000).

#### INDIA

## STUDIES FOR ONSITE DISPOSAL OF WASTE FROM DECOMMISSIONING/REVAMPING OF NUCLEAR FACILITIES AND NPPS IN INDIA

#### S. Kumar, P.M. Satya Sai, S. Manohar, R.R. Rakesh,

Nuclear Recycle Group, Bhabha Atomic Research Centre, Trombay, Mumbai

#### Abstract

The Indian nuclear power programme is about five decades old, with many nuclear facilities that are aging. Planning is underway for decommissioning of these facilities. Approval for extension of the operational life of some of these facilities has been obtained from regulatory bodies after refurbishment of their critical systems. However, their decommissioning in the near future is imminent. These refurbishment activities have resulted in accumulation of contaminated equipment and structural material which require special consideration. Experiences gained during some of these partial decommissioning/refurbishing activities are described in this report.

This report describes the experiences gained for management of LILW generated during

- Enmasse coolant channel replacement (ECCR) campaign for RAPS-II, MAPS- I and II (220 MWe- PHWRs),
- Decommissioning/disposal of contaminated equipment from Waste Immobilization Plants
- Decommissioning of thorium processing radio-chemical plant.

Development of processes and technologies addressing decommissioning, dismantling, compaction, packaging and storage of the retired equipment is an ongoing activity.

#### 1. INTRODUCTION

The Indian nuclear power programme is comprised of fuel production, power/research reactors, fuel reprocessing, research and development, isotope production and radioactive waste management. In India, the radioactive waste management programme for segregation, treatment /conditioning, and storage/disposal of waste was also developed from the inception of the nuclear programme, along with related research and development activities. Near surface disposal facilities are collocated with the nuclear facilities, which are spread throughout the country. As the nuclear plants and facilities age, major programmes have been initiated in the recent past for augmentation, modifications and upgrading which necessitated partial decommissioning. The experience gained over the last five decades for management of waste generated during operation and maintenance activities has been successfully applied to the management of LILW generated from such decommissioning activities as per regulatory requirements [1].

## 2. CASE STUDIES

Typical case studies of the experience are presented in this report.

## 2.1. Enmasse coolant channel replacement (ECCR) campaign for PHWRs

A major portion of Indian nuclear power programme consists of Pressurised Heavy Water Reactors (PHWR). The Rajasthan Atomic Power Station is located in the north-west region of India and unit-II of these twin units of 220MWe each had seen 8.5 full power years of operation. Enmasse coolant channel disposal campaign of unit-2 of Rajasthan Atomic Power Station (RAPS-II) was carried out during April - September, 1996. The Madras Atomic Power Station (MAPS), located in the Southern region of India, is also a twin unit station of 220 MWe each. Replacement of about 300 coolant channels of each unit with zirconium niobium tubes was found necessary due to various factors like induced neutron/hydride embrittlement, creep/growth, fatigue and corrosion. The MAPS - 1 of the station had seen 10.1 full power years (FPYs) and MAPS –II had seen about 9.5 full power years of operation before the replacement was carried out. Removal of coolant channels was carried out for MAPS- II during May –July 2002 and for MAPS – I during January- March 2005. Enmasse coolant channel disposal campaign of unit-2 of Rajasthan Atomic Power Station (RAPS-II) was carried out during 1996 and the results were presented in the earlier report [2].

Radioactive waste management during these three ECCR campaigns was planned and carried out safely. The disposal of the waste materials required meticulous planning and concerted efforts due to high radiation fields, and large quantities and odd dimensions of the components requiring creation of additional facilities for their handling, transport, cutting, sizing, disposal and conditioning. For each operation, new technologies were developed and the existing technologies were improved upon.



- 1 coolant tube
- 2 calandria tube
- 3 end fitting body
- 4 end fitting liner
- 5 3 grooved balled joint
- 6 calandria tube sheet
- 7 end shield structure

8 end shield water passage

- 9 fuel bundle
- 10 shielded plug
- 11 flow control tube
- 12 closure plug
- 13 sealing plate
- 14 feeder connection
- 15 split shield tube
- 16 air gap
- 17 quarter springs
- 18 latch
- 19 calanderia stop
- 20 end fitting bearings
- 21 end stop and anti torque device

Fig. 1. A schematic sketch of the coolant channel.

## 2.2. Quantities and radio assay of the materials involved

The main components of this coolant channel that needed disposal were end fittings (EF), pressure tubes (PT), garter springs. Material of construction of different components is given in Table 1.

Item	Material of construction					
Pressure tubes	Ni free Zircalloy – 2					
End fittings	S.S-403					
Garter springs	Zr-Cu-Ni alloy					
Shield plugs	S.S – 410 A					
Feeder pipes	C.S ASTM A 106 Gr B					

Table 1. Material of construction of coolant channel

Radionuclides of major concern from consideration of safety which were present in these components are listed in Table 2.

Radionuclide	Half life	Major energies	Production mode
Co-60	5.3 Y	1.17 (100%)	$\frac{\text{Co-59}(n,\gamma)}{100}$
		1.33 (100%)	N1-60 (n,p)
Co-58	71 D	0.11 (99%)	Ni-58 (n,p)
Mn-54	301 D	0.84 (100%)	Fe-54 $(n,p)$
			WIII-34 (II,2II)
Fe-59	45 D	1.10 (56%)	Fe-58 $(n,\gamma)$
			CO-39 (II,p)
Cr-51	27.8 D	0.32 (9%)	Cr-50 (n,γ)
Sb-125	2.7 Y	0.60 (100%)	Sn-124 $(n,\gamma)$
			Sn-125 p_emission
Zr-95	65.5 D	0.72 (49%)	Zr-94 (n,γ)
Nb-95	35.5 D	0.76 (100%)	Zr-95 $\beta$ _emission

Table 2. Major radionuclides of concern in coolant channel

Inventory of these wastes disposed in all the three campaigns along with their physical and radiological characteristics are given in Table-III.

No	ITEM	RAPS-II	MAPS-II	MAPS-I
1	Duration of Campaign	25.04.96-10.09.96	06.05.02-27.07.02	05.01.05- 18.03.05
		(139 days)	(83 days)	(72 days)
2	Full power years (FPY)	8	9.5	10.1
3	Cooling Period	2 years +	EF: 120-147 days PT: 165-205 days	503 days
4	Waste Inventory			
a	Pressure tubes	296 Nos	290 Nos.	298 Nos
		0.86-6.19 Gy/h	~70 Gy/h	>10 Gy/h
b	End fittings	608 Nos	600 Nos 8 Gy/h	594 Nos
		0.72-6.70 Gy/h		>10Gy/h
c	Garter Springs	598 Nos	600 Nos >10Gy/h	594 Nos
		0.1-0.15 Gy/h		10 Gy/h
5	No. of tile holes consumed	105	90	POOL + 52
6	Total tonnage of material handled	21,174	21,764	27,000
7	Total activity handled	7.4E+05 GBq	1.4E+07 GBq	5.4E+06 GBq
8	Person mSv expenditure	17.27	50.267	4.50

The irradiated reactor components have a large inventory of activation and corrosion product radionuclides. The longest half life amongst these radionuclides is that of <sup>60</sup>Co (5.3 years). Various nuclides after 30 days of cooling were found to be as <sup>95</sup>Nb (T<sup>1</sup>/<sub>2</sub>-35d), <sup>95</sup>Zr (T<sup>1</sup>/<sub>2</sub>-65.5d), <sup>58</sup>Co (T<sup>1</sup>/<sub>2</sub>-71.3d, <sup>60</sup>Co (T<sup>1</sup>/<sub>2</sub>-5.263y), <sup>59</sup>Fe (T<sup>1</sup>/<sub>2</sub>-45.6d), <sup>125</sup>Sb (T<sup>1</sup>/<sub>2</sub>-2.71y), <sup>113</sup>Sn (T<sup>1</sup>/<sub>2</sub>-115d) and <sup>123</sup>Sn (T<sup>1</sup>/<sub>2</sub>-125d).

## Radioactive waste disposal schemes

At RAPS, the pressure tubes were cut into two pieces and disposed into tile holes. The cutting machine utilized a power saw housed in a shielded enclosure to serve as a hot cell for cutting/transfer of the pressure tubes.

During the campaign for MAPS –II in 2002, two coolant tube cutting systems were used for size reduction of the coolant tubes for disposal into tile holes. One of these machines was utilised for size reduction of coolant tubes used at RAPS in 1996. This machine utilizes a

power saw housed in a shielded enclosure to serve as a hot cell for cutting/transfer of the coolant tubes. As an improvement of this machine, another chipless cutting machine was designed and used in the campaign for MAPS-II in 2002. This machine achieves cutting via a rotary orbital tool. Both the motion i.e. revolution of cutter and feeding are achieved by a common drive. Each revolution of the cutter causes fine indentations of the tube and cutting is performed by progressive indentation and tearing of the tube. The end fittings and garter springs were disposed in tile holes [3].

For the disposal from MAPS-I, further improvements on the earlier campaigns were planned and implemented as follows:

(a) As an alternative to cutting and disposal of pressure tubes, underwater storage of the pressure tubes was performed. This allows for size reduction and recovery and reprocessing of zirconium at a later date. A process for recovery and reprocessing of zirconium from these pressure tubes is under development. A view of the under water storage pool for pressure tubes is given in Fig 2.



Fig 2. Water storage pool for Irradiated Pressure Tube from MAPS-I built at Central Waste Management Facility, Kalpakkam.

(b) In the previous campaigns, the end fittings were taken out of the cask into open air and disposed of in the tile holes by erecting a temporary containment structure around the disposal spot. In the latest campaign, the end fittings were transported in a cask with door/valve on both sides. At the disposal site, the end fitting were grabbed by a special tool and disposed by lowering into the tile hole, without bringing it into open air, thus achieving a significant reduction in radiation exposure to personnel.

End fittings were disposed of into tile-holes, using an improved scheme described earlier. The other materials such as garter springs and seal plug disc were disposed of in RCC trenches. This was followed by the closure of these tile holes, after reducing the surface radiation fields to safe (prescribed) limits. The design of the doughnut shape facilitated alignment of the disposal cask with each opening to achieve the unobstructed passage of the pressure tube piece or end fitting into the tile hole, and to provide shielding to the personnel during disposal. It also ensured the vertical standing position of the individual tube pieces or end fittings in the tile hole before and after grouting operations. The tube guide assembly can be placed on top of the tile hole and is provided with 16 openings. All of the openings are

provided with a shielded gate or plug in the top cover and a 3 meter long pipe at the bottom face. This results in an easy disposal operation with minimum exposure to the operating personnel.

To validate the estimate of radioactivity, a typical pressure tube was removed from the reactor core and subjected to radio assay and metallurgical examination. The possibility of alpha ( $\alpha$ ) emitting nuclides on the inner surface of the coolant channels and end-filling and also the presence of long lived beta ( $\beta$ ) emitters like C-14 was also investigated. No significant alpha ( $\alpha$ ) contamination was reported. The Carbon-14 deposition was also reported to be insignificant.

# 2.3. Experience in decontamination/disposal of contaminated equipment in Waste Immobilization Plants.

Waste Immobilization Plants (WIP) at Tarapur and Trombay are in operation for vitrification of HLW generated during fuel reprocessing. Continuous operation of these facilities demands decontamination and replacement of certain equipment and systems, which need to be subsequently disposed of safely. Suitable technologies and material handling systems are in place to address such requirements. In addition, due to the complexity of operations involving high temperatures, corrosive chemicals and radiation fields, some off-normal situations have to be handled in a systematic way to maintain safety.

Described below are some typical decontamination/decommissioning experiences for the vitrification facilities.

## Replacement of process pot and susceptor from the WIP- Trombay

At WIP Trombay, high level waste contains significant amount of sulphate and is vitrified in barium borosilicate glass matrix in induction heated metallic melter [4]. The provision has been made to replace the process pot (PP) and susceptors remotely after completion of predetermined operating cycles. A schematic of the induction heating furnace assembly is shown in Fig. 3. However, during one such replacement, it was noticed that the process pot was entangled with the susceptor due to deformation and dimensional changes. Thereafter, combined removal of process pot and susceptor was attempted but the assemblies could not come out of the furnace. Efforts to separate the process pot from the susceptor by heating the furnace and pulling out the PP mechanically were unsuccessful.

Efforts were made to remove the process pot from the furnace after being disconnected from fill heads. Even after repeated attempts, the entangled processes pot and susceptors could not be removed. It was then decided to cut the furnace top plates remotely and remove the entangled process pot and susceptors along with the guide plates.

A special cutting device consisting of a miniature motor and appropriate guiding arrangement with grinder was designed and fabricated. The device was used to cut the furnace top plates, which allowed the equipment to be removed successfully. The process pot and susceptor were then overpacked in a stainless steel over pack and welded remotely prior to disposal. The development of specific devices and the experience gained in decontamination and handling of such equipment from cells will go a long way in management of waste arising during decommissioning activities



Fig. 3. A schematic of induction heating furnace assembly.

# Decontamination/decommissioning of air lock cell and process cell, Tarapur

At WIP Tarapur, the air lock cell is a 4m X 11m cell housing two numbers of shielded boxes and one fume hood. The fume hood houses the gas sampling system. There was an incident of blow back of activity into the shielded boxes and on the floor below. This resulted in a general background of 2-5 mGy/h. A few hotspots of 700-1000 mGy/h were also detected. Decontamination of epoxy painted floor with water and mild nitric acid resulted into little reduction in the radiation field, indicating the fixed nature of contamination. It was decided to dismantle the shielded boxes and the fume hood followed by chipping of the floor to a depth of 100 to 150 mm and reconstructing the surface. The quantity of waste generated during this campaign was about 50 tonnes and a 50 person mSv expenditure was incurred. Another important aspect of this D and D campaign was to recover all the useful and expensive items such as MSMs, weighing balances etc. for reuse.

Revamping activities were also begun in the process cell, which involved refurbishment of its in-cell equipment including metallic melters and remote handling and viewing equipment; providing additional systems and facilities. These activities are estimated to be about 50% equivalent to total decommissioning efforts for such plants. The details of chemical and radiochemical characteristics of waste and quantities are given in Table 4.

Type of Waste	Quantity	Radiation field
High Level Solids:		
Metallic waste		
Process pots (High Ni-Cr alloy (60/30/10))	4 Nos.( 300 Kg.)	1-4 Gy/h.
		-
Low level:		
Metallic susceptors	4 Nos.( 300 Kg.)	1-5 mGy/h.
Assorted solid waste	$5 \text{ m}^3$	1-2 mGy/h.
		-
Liquids:		
High level (Decontamination liquids)	$1.5 \text{ m}^3$	1-2 MBq/liter.

 Table 4. Characterization of waste during process cell revamping

## 2.3. Decommissioning of thorium processing facility

The thorium plant at Trombay was commissioned in 1955 to process thorium hydroxide concentrate into thorium nitrate. Due to ageing, structural weakness, extensive corrosion and build up of radiation dose on the process equipment, it was decided to decommission the plant in the year 2000. The main chemical processing steps consist of sequential dissolution of thorium concentrate in acids and production of dry solid thorium nitrate by evaporation and drying.

The main plant had a volume of 11 000 m<sup>3</sup> (42m x 21m x 12.4m) and Solvent Extraction Plant had a volume of 1000 m<sup>3</sup> (20m x 10m x 4.9m). The radiation fields before decommissioning in the various areas were 5-300 mGy/h. The floor chipping samples from various plant areas indicated contamination due to <sup>228</sup>Ra to be 0.4 -90 Bq/g ,due to <sup>228</sup>Th to be 0.4 -99 Bq/g and due to <sup>226</sup>Ra to be 0.2-4 Bq/g. This facility was decommissioned and the radiation fields were reduced to background levels.

The decommissioning was planned so as to minimize both the radiation exposure to working personnel and the generation of radioactive waste. The steps taken were decontamination of tanks and equipments, removal of the tanks and equipment from the plant and subsequent removal of civil structures, segregation of waste, and disposal of radioactive waste in dykes.

The radioactive solid waste about 2150 m<sup>3</sup> consisting mainly of 70% concrete and 30% as metallic waste was disposed in the dykes of size  $25m \times 25m \times 3.5m$  and  $33m \times 37m \times 3.5m$ . During the decommissioning about 300 m<sup>3</sup> of contaminated soil excavated from three main drains also was disposed in the dykes. The dose on the surface of the dykes is below 0.01 mGy/h.

A total of about 3465 man-days were required for decommissioning with a radiation exposure of 122 person-mSv [5]. A systematic scheme is being planned for safe closure of this disposal site at Trombay. The plant has handled natural radioactivity and the experience gained during decommissioning of a radiochemical plant is of great importance.

# 3. SAFETY ASSESSMENT OF DISPOSAL FACILITIES

To ensure that the disposal of the waste from this operation does not impose any undue impact on the environment, a safety analysis was performed. Considering the nature and long term behaviour of the waste (reactor components that are non-leachable and non-dispersible, and immobilized in metallic matrix), and the type of near surface disposal facilities under use in India, a decision to dispose of this waste in a co-located near surface disposal facility was considered to be safe.

In the case of WIP decommissioning waste the ILW/HLW decontamination liquid waste was either concentrated and recycled, or decontaminated via ion-exchange treatment and recycled. Only LLW was sent to the low level waste treatment plant for treatment and disposal.

The high level solids were packaged and stored for future disposal, while assorted low level solid packages meeting disposal criteria were placed in near surface disposal.

## 4. TECHNOLOGY/EQUIPMENT DEVELOPMENT FOR DECONTAMINATION/ DECOMMISSIONING

Decommissioning activities require development of tools and technology to handle large quantity of solid waste with varying contamination levels. Decontamination and surveillance form an integral part of decommissioning activities to minimize radiation exposure to working personnel. Remote mechanical handling devices are required to be developed to achieve these objectives. Some of the major efforts in this direction are briefly described.

## 4.1. Remote inspection device (RID)

Remote Inspection Device (RID) is equipment developed for in-service surveillance/inspection of underground waste storage tanks and their containment structure. It consists of a radio controlled mobile vehicle that carries the surveillance equipment/ instruments and a remote station for control of the vehicle and display of the data collected by the vehicle. The basic mission of the vehicle is to perform videography of the tank farm, and air sampling, radiation monitoring and swab collection of tank surface. A photograph of RID is presented in Fig. 4.

The radio controlled mobile vehicle of RID consists of a power/steering module, payload module comprising of the instruments, articulated arm and camera module as per mission requirements. The battery pack is designed for a one-hour mission.



Fig 4. Remote Inspection Device (RID) developed for waste storage farm.

The RID unit has an overall size of 600 mm x 500 mm and is about 800 mm in height. This equipment could also find extensive use in radiochemical cells and other nuclear facilities for In-service Inspection (ISI).

## 4.2. Servo robot for decommissioning (SRD)

Servo Robot for Decommissioning (SRD) is a six axes servo controlled gantry robot with 25 Kg payload capacity of the arm. A photograph of SRD is presented in Fig. 5. SRD is intended for volume reduction of components/ equipments declared as metallic waste in nuclear facilities.



Fig 5. Servo robot for decommissioning.

# 4.3. Remotely operated forklift

A radio frequency controlled, battery operated forklift has been designed and developed for remote handling of low and intermediate waste stored in drums. A photograph of a fork lift is presented in Fig.6. All the operations such as lifting, moving, tilting, steering, braking, grappling of drum are carried out remotely with the help of radio frequency signals from control panel located at a distance of 300 meters. The forklift is provided with a fully articulating drum-grabbing unit that can pick up and place standard 200 litres of drum at desired locations.



Fig 6. Remotely operated forklift.

This remotely operated forklift is extremely useful for reducing radiation exposure during handling of low-active waste drums in near surface disposal facilities and also in waste assaying areas.

# 4.4. High capacity mobile cementation unit (MCU)

A mobile cementation unit has been developed for immobilization of radioactive low level liquid waste, and is comprised of a cement waste mixing unit, cement handling unit, control console with operator's cabin and ventilation system. The unit is assembled inside a specially designed, fully enclosed 2.4 m x 2.9m x 6.1m dry freight container with removable top hood

trailer and can be transported between sites in a ready-to-use condition. A photograph of MCU is presented in Fig. 7. The system is designed as a fully automatic unit and can be operated from a central control cabin through PC/PLC or through push buttons. The unit is designed for a waste flow rate of 600 l/hr and cement feed rate of 900 kg/hr. The waste form has been shown to meet waste acceptance criteria. The unit can handle waste slurry with 1% suspended solids. The operating experience with mobile cement unit for management of radioactive sludges is quite encouraging. This experience will be utilized for handling large amounts of radioactive waste generated during decommissioning requiring in-situ immobilization.



Fig 7. High capacity mobile cementation unit.

## 4.5. Laser cutting system for pressure tubes

In order to demonstrate the technology for laser cutting of metallic components, seven pressure tubes were cut into two pieces and disposed in one tile hole during ECCR campaign for MAPS-II in 2005. The laser used was Nd-YAG (Neodymium-Yttrium Aluminium Garnet). Cutting assist gas was either argon or nitrogen. The system consists of a power supply unit, chilling unit, laser generator, fibre optical cable to carry laser beam to cutting tool, gas cylinders and a cutting tool.

The laser cutting tool was installed in the cutting chamber of the chipless cutting machine. The fibre optic cable with gas sheath enters the cutting machine chamber in one of the existing plugs. The cutting tool rotates over the tube by 360° by using a stepper motor fixed over a stand. A few microns thick material was vaporised by laser and blown away inside the tube by assist gas and instantly solidified. The gases from the chamber were exhausted through a HEPA filter bank. Other units were housed outside the cutting chamber in inactive areas, and were easily maintainable. Inactive mock-up testing operation was carried out with zircalloy tubes and carbon steal tubes to optimise the parameters, before the system was put to active duty.

By cutting seven pressure tubes, it was seen that the cutting was smoother compared to mechanical cutting with no chips or aerosols generated. As the material is vaporised and instantly solidified adhering to the tube itself, no secondary waste or air contamination was observed. No maintenance problems were encountered unlike in mechanical cutting, resulting in considerable reduction of downtime and personnel exposures.

#### 5. CONCLUSION

Replacement of coolant channels in PHWRs, which amounts to partial decommissioning of the reactor, was carried out in India on three occasions during 1996, 2002 and 2005. The wastes generated during these operations were of a varied nature, the majority being highly radioactive and heavy in nature. Handling and disposal of such wastes required the design and development of suitable technologies and equipment, meticulous planning, formulation of procedures and training of the operating personnel. This was achieved with improvements being incorporated in each operation based on the feed back from earlier operations. The development of specific gadgets and the experience gained in decontamination and handling of such equipment from cells will go a long way in management of waste arising during decommissioning activities. Valuable experience has been gained in disposal of large quantities of waste in dykes generated during decommissioning of a radiochemical plant handling natural radioactivity. As a result, technology for the management of decommissioning of reactors and other facilities is getting established in the Indian nuclear power programme. Operations involving partial decommissioning of other radioactive facilities and management of the wastes were carried out successfully with minimum exposures. Technologies and equipment which aid management of such waste are under continuous development in Indian nuclear power programme.

#### ACKNOWLEDGEMENT

The experience gained in management of en masse coolant channel replacement campaigns at Rajasthan and Kalpakkam is the result of the dedicated efforts of a large number of technical and scientific personnel at the two sites. Management of highly contaminated materials during operations and refurbishment activities in waste immobilization plants at Trombay and Tarapur was successfully accomplished by coordinated efforts of scientists from different disciplines. The authors gratefully acknowledge the valuable contribution of their colleagues in Nuclear Recycle Group, BARC. The authors are also grateful to Shri Kanwar Raj, Head, Waste Management Division, BARC for his continuous support, encourage and guidance.

#### REFERENCES

- [1] AERB safety manual, Decommissioning of nuclear facilities AERB / SM / DECOM-1 India (1998).
- [2] ALI, S.S, et al. "Management of low and intermediate waste from pressurized heavy water reactors enmasse retubing and high level waste immobilization revamping operations in Inda", 2<sup>nd</sup> IAEA-CRP meeting on disposal aspects of low and intermediate level decommissioning waste, Argentina (2004).
- [3] DHANAVADE, P.D., et al, "Safety of radioactive waste disposal during enmasse coolant channel replacement of Indian PHWR" International conference on the safety of radioactive waste disposal, Tokyo, Japan (2005).
- [4] RAJ KANWAR, et al, "Commissioning and operation of high level radioactive waste vitrification and storage facilities" Nuclear fuel cycle technologies-closing the fuel cycle, 14<sup>th</sup> Annual Conference of Indian Nuclear Society INSAC, India (2003).
- [5] SAVANT, P.B., et al, "Health physics experience on the decommissioning of thorium plant of I.R.E.Ltd at Trombay", BARC/2004/I/011, India (2004).

#### **REPUBLIC OF KOREA**

## A STUDY ON THE DISPOSAL CONCEPT AND PACKAGING OF LOW- AND INTERMEDIATE-LEVEL DECOMMISSIONING WASTE IN KOREA

## C-L. Kim, J.B. Park, J.W. Park

Korea Hydro and Nuclear Power Co., Ltd., (KHNP) Nuclear Environment Technology Institute (NETEC), Daejeon

### Abstract

The decommissioning project for two TRIGA type research reactors in Korea was started in 1997. A decommissioning plan including disposal method of decommissioning waste should be submitted to the regulatory authority for licensing of nuclear facilities. A national radioactive waste repository will be operational from 2008. For the successful implementation of the national disposal programme, the concept and packaging for the disposal of domestic decommissioning waste needs to be investigated. In this study, a survey has been conducted on the status of decommissioning activities of nuclear facilities. Waste packaging and source-term considerations along with decommissioning waste characterization are investigated. Gas generation by corrosion of metals both in the waste itself and in its associated packaging are assessed for all types of waste forms and packages being used. Release of gas from disposal vaults under repository conditions is assessed. Finally, conceptual design and preliminary safety assessment for the proposed disposal facility containing decommissioning waste are also performed.

#### 1. INTRODUCTION

The safe management of radioactive waste is a national task required for sustainable generation of nuclear power and for energy self-reliance in Korea. Nuclear power generation was first introduced in 1978 in Korea. Since then, the rapid growth in nuclear power supply has been remarkable. As of January 2006, a total of 20 nuclear power units are in operation with installed capacity of about 17 GWe, and six units are under construction. Currently, the electricity generation from nuclear power plants accounts for about 40% of total electricity generation. A plan for the future nuclear power plants is ambitious. Such a large nuclear power generation programme has produced a significant amount of radioactive waste and will generate more in the future.

Apart from the nuclear power generation programme, there are three research reactors, two TRIGA types and one multi-purpose research reactor in Korea. Both of these two TRIGA type research reactors were shut down in 1995, and the decommissioning project for them was started in January 1997. Meanwhile, the first commercial nuclear power plant, the Kori unit 1 (587 MWe, PWR) and Wolsong unit 1 (679 MWe, PHWR) will apply for life-extension after 2007 and 2013, respectively.

KHNP-NETEC is responsible for the disposal of all radioactive waste in Korea. A national radioactive waste repository will be operational beginning in 2008. For the successful implementation of the national disposal programme, the concept and packaging for the disposal of the domestic decommissioning waste needs to be investigated. This paper describes a plan for a three year study on the disposal concepts and major issues related to the decommissioning waste to be considered in the design and safety assessment of the Korean repository. In addition, the status of decommissioning activities of nuclear facilities in Korea is presented.

This study was initiated from Sept. 2002. In the first year of the study, a survey on the status of decommissioning activities of nuclear facilities has been conducted. In 2003, waste packaging and source-term consideration along with decommissioning waste characterization are investigated. In 2004, gas generation by corrosion of metals both in the waste itself and in its associated packaging are assessed for all types of waste forms and packages being used. Radionuclide release as a gaseous form from waste packages and disposal vaults under repository conditions are assessed. Finally, conceptual design and preliminary safety assessment for the proposed disposal facility containing decommissioning waste are also performed in 2005.

2. STATUS OF DECOMMISSIONING ACTIVITIES OF NUCLEAR FACILITIES IN KOREA

There are three research reactors in Korea. Korea Research Reactor-1(KRR-1), the first research reactor in Korea (TRIGA Mark-II, 250kW), and KRR-2, the second one (TRIGA Mark-III, 2000 kW) started their operation in 1962 and 1972, respectively, both of them were shut down in 1995. A new multi-purpose research reactor named HANARO (High-flux Advanced Neutron Application Reactor) in Daejeon began its operation

The decommissioning project for these two TRIGA type research reactors was started in January 1997. When it has been decided to shut down a nuclear facility, the operator shall submit an application for permission to decommission the facility for approval by the regulatory authority, together with the proposed final decommissioning plan including:

- Decommissioning method and schedule
- Decontamination method of the contaminated materials
- Treatment and disposal method for the radioactive wastes
- Countermeasure for the protection from the radiation damage
- Environmental impact assessments and its countermeasure
- Other matters required by the MOST(Ministry Of Science and Technology)

Before a site may be released for unrestricted use, a survey shall be performed to demonstrate that the end point conditions, as established by the regulatory body, have been met. If a site cannot be released for unrestricted use, appropriate control shall be maintained to ensure protection of human health and the environment.

For KRR-1 & 2, the decommissioning plan documents for licensing including the environmental impact assessment were prepared and submitted to the MOST in Dec. 1998. After regulatory review by the KINS (Korea Institute of Nuclear Safety), the decommissioning plan was approved in Nov. 2000. The decontamination and decommissioning works for KRR-1 & 2 are to be performed until the end of 2007. A tentative schedule of the decommissioning project for KRR-1 &2 is shown in Figure 1.

Year	,07	,08	,00	,00	,01	,02	,03	,04	,05	,06	,07	,09
Activity	97	90	99	00	01	02	03	04	05	00	07	00
• D&D plan												
D&D design document												
Environmental assessment												
Licensing		_										
Spent fuel transportation												
D&D Works												
Site restoration												
Radiation monitoring												
R&D demonstration												
Waste treatment/storage												
Waste transportation												
Final radiation Survey												

## Fig. 1. Decommissioning schedule of KRR-1 & 2

According to the decommissioning plan proposed by KHNP, dismantling and demolition of nuclear power plants will be conducted after safe storage of  $5 \sim 10$  years, subsequent to the decay of radioactivity and system decontamination with consideration of dismantling two adjacent units together. Dismantling and decontamination techniques that minimize waste arisings and airborne contamination will be chosen. Decommissioning activities such as decontamination, cutting and handling of large equipment and the progressive dismantling or removal of some existing safety systems have the potential for creating new hazards. Proven techniques and equipment are now available to dismantle nuclear facilities safely. The safety impacts of the decommissioning activities will be assessed and managed so that these hazards are mitigated.

As an effort to develop the enhanced decontamination and decommissioning technologies, Korea is participating in the International Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects of the OECD Nuclear Energy Agency. KHNP is setting aside the required funds for the implementation of decommissioning projects for nuclear power plants, which by law cannot be used for any purposes other than decommissioning.

## 3. TECHNICAL STANDARDS AND CRITERIA RELATED TO WASTE DISPOSAL

Two criteria, performance criteria for the repository and waste acceptance criteria, which are closely related to the disposal concept and waste packaging for decommissioning waste, are briefly introduced.

## 3.1. Performance Criteria (MOST NOTICE 2005-17) [1]

These criteria specify the details necessary for assessing the performance of a repository for the LLW disposal. Some articles of great importance are as follows:

— Article 6 (Performance objective) The predicted radiological risk to any individual in a critical group that is assumed to be located at a time and place where the risks are likely

to the greatest from LILW repository, shall not exceed the risk criteria of  $10^{-6}$ /yr based on aggregated dose/likelihood approach into dose constraint of 0.1 mSv/yr based on disaggregated dose/likelihood approach.

- Article 7 (Time scale of performance assessment) The period demonstrating compliance with the individual risk requirement need not exceed 1000 years. When the predicted risk dose does not show peak value(s) in 1000 years, there must be reasonable arguments that beyond 1000 years the rate of radionuclide release into the environment will not suddenly and dramatically increase and that acute radiological risks will not be encountered by future generations.
- --- Article 10 (Estimate of individual risk-safety assessment) The calculation of individual risk shall be made either by annual individual effective dose calculated as the output from deterministic pathway analysis or by arithmetic mean value of annual individual effective dose from the distribution of individual effective doses in a year calculated as the output from probabilistic analysis. The risk conversion factor of  $5 \times 10^{-2}$  per Sv is recommended.

# 3.2. Acceptance Criteria for Waste Packages (MOST NOTICE 2005-18) [2]

Acceptance criteria are to confirm the integrity of the solidified and packaged waste to be disposed of with both generic site condition and the Korean-specific repository design conditions. Several important articles are as follows:

- Article 6 (Limitation of activity) Radioactive wastes shall be classified according to the activity. The concentration of each radionuclide, total radioactivity, and the contents in the waste shall meet the requirements for the radioactive waste disposal.
- Article 10 (Type of the waste) The type of waste shall meet the following requirements:
  - The waste shall be solidified to ensure the handling and post closure safety;
  - The waste shall be solidified uniformly.
- Article 11 (Conditions of the package) The waste package shall meet the following requirements:
  - All waste forms shall be packaged with incombustible material, and no defect shall be found by the visual inspection;
  - The structural strength of the waste package shall be maintained under the accidental situation during the handling and transport, and it can overcome the pressure growth caused by gas generation;
  - The type, volume, and weight of the waste package shall meet the requirements of handling and transport of radioactive material.
- Article 13 (Free-standing liquid) Free-standing liquid in the waste package shall be minimized as little as possible, and the volume of the free-standing liquid shall not exceed the 0.5 volume percent of the package.
- Article 14 (Corrosion) The waste forms containing the corrosive material shall reduce the corrosive properties and be packaged to resist the corrosion.

# 4. RADIOACTIVE WASTE PACKAGING AND WASTE SOURCE TERM

# 4.1. Waste Packages

For the research reactors, most solid wastes, except some parts of the reactor structure, have low-level radioactivity. Intermediate-level waste from the reactor structures will be stored in a

shielded container (TIF cask). The rest of the low-level solid waste will be packed in a 4 m<sup>3</sup> ISO container. For that purpose, 50 containers were specially designed and fabricated in the year 2001. The waste will then be temporarily stored in the KRR-2 reactor hall until a national LILW repository is operational. As for the decommissioning waste of nuclear power plants, waste packages of 200 $\ell$  drums are considered for disposal at the national LILW repository.

## 4.2. Waste Volume and Source Term

Low- and intermediate-level radioactive wastes include both the operational waste generated from NPPs or related industries and the decommissioning waste. The total waste volume to be disposed of in national the LILW repository is 800 000 drums, including both the operational waste (400 000 drums) and the decommissioning waste (400 000 drums).

The disposal amount of both operational and decommissioning waste is divided by individual phase in which 100 000 drums of operational waste and 100 000 drums of the decommissioning waste will be disposed of in the national LILW repository. The expected radionuclide inventories of the decommissioning waste are listed in Table 1.

Radionuclide	Inventory (Bq)
Co-60	3.09E+16
Ni-59	1.02E+13
Ni-63	1.62E+15
Sr-90	1.07E+12
Nb-94	6.96E+10
Cs-137	1.21E+15

Table 1. Radionuclide inventor	y of the decommissioning	waste (400,000 drums)
		,

# 4.3. Waste Disposal System

KHNP-NETEC is considering the two alternative disposal methods, the rock cavern and the engineered vault disposal for a planned LILW repository, and a preferred type will be determined in consideration of site conditions. Conceptual design studies and preliminary safety assessments for both rock-cavern type and engineered vault type disposal facilities were completed in 1993 and 1999, respectively. In the rock cavern disposal facility, the decommissioned waste containers will be placed into caverns for low level waste (LLW). Three types of caverns for low-level waste will be constructed according to waste types: LLW I cavern for dry active wastes; LLW II caverns (two caverns) for DAW and concentrated wastes; and LLW III cavern for spent resin, spent filter and concentrated wastes. The LLW will be handled by forklift in these caverns. The engineered vault disposal facility consists of three types of vault depending on the durability and/or size of waste packages: Vault I (waste packages with long durability and backfilled with gravel); Vault II (standard size waste packages with short durability and grouted with cement mortar); and Vault III (large size waste packages with short durability and grouted with cement mortar). The capacity of a vault will be 5000 drums based on 200ℓ drum. The disposal vault is covered with a mobile roof during the waste package loading. The mobile roof equipped with waste package handling crane can be moved to the next vault for another loading operation. The final disposal cover will be constructed when the disposal vaults in a disposal area are completely filled. The final cover consists of a multi-laver system to ensure low percolation, water drain, and intrusion resistance.

## 5. GAS GENERATION/MIGRATION OF THE DECOMMISSIONING WASTE

#### 5.1. Evaluation of Gas Generation from Decommissioning Waste and Waste Packages

The waste acceptance criteria [2] for waste packages promulgated by Korean regulatory body require that not only the waste packages should be able to overcome the pressure growth caused by gas generation, but also the performance of repository and integrity of waste packages should not be compromised by gas generation from radiolysis, or biological and chemical reaction. For the assessment of the expected gas production in the decommissioning waste packages, corrosion of metals and microbial activity of organic materials were considered as the principal gas generation mechanisms.

The principal gases formed by these mechanisms are hydrogen, carbon dioxide and methane. Gas generation from three types of Dry Active Waste (DAW) packages, namely 200-L steel drum (Type 1), 350-L repackaged drum (Type 2), and concrete-shielded steel drum (Type 3) is assessed for a repository environment expected in near-surface vault type disposal facility. The GAMMON programme [3] is applied to estimate the cumulative amounts and generation rates of major bulk gases with time over a period of one thousand years under a given set of proposed disposal vault condition. The assumed condition of disposal vault is summarized as follows:

- (1) Disposal vault dimension: 20m W×20m D×8.1m H
- (2) Backfilling strategies of the vaults: backfilling with gravel for long-term packages such as concrete-shielded drum, and grouting with cement for short-term packages such as steel drum
- (3) Number of packages per vault: 4000 packages for Type 2 and 5000 for the other type of packages
- (4) Near-field condition: completely anaerobic and resaturated with surface water or groundwater at the time of closure of the vaults, buffered at pH 9 for gravel-backfilled vault and at pH 12 for cement-grouted vault.

The results of the GAMMON calculations to determine cumulative amounts of major bulk gases generated for individual drum types show that a cumulative total of 250~670 mol per package will be generated during  $10^3$  years after closure. In order to determine the effects of input parameters on the calculation results, a sensitivity analysis is also conducted for the Type 1 package. The key parameter affecting gas generation is the pH of vault porewater. This parameter has a significant effect both on the rate of anaerobic metal corrosion and on the rate of microbial degradation of cellulosic wastes.

## 5.2. Evaluation of gas migration in a disposal facility

In order to estimate pressure build-up caused by gases generated from metal corrosion and microbial degradation inside the engineered barrier structure under repository condition, a simple analytical assessment is performed. The principle objective of this assessment is to obtain some indication of the potential for significant over-pressurization within the disposal vault.

In the analysis, mass conservation within vault volume at time and Darcy's law for a compressible flow to represent the gas flux through the barrier are applied. For the purpose of this assessment, both the peak total gas generation rate used for Type 1 package. It is assumed that the gas can be treated as hydrogen with a viscosity of 8.5E-6 Pa-sec. Using these data, the

internal pressure within the vault is calculated with various gas permeabilities

 $(1 \times 10^{-11} \times 10^{-15} \text{ m}^2)$ . The internal pressure in a vault comprising 5,000 DAW drums is calculated to be 0.101~0.106 MPa after 1,000 years. Therefore, it is expected that over-pressurization within the disposal vault caused by gas production will not be occurred.

# 6. PRELIMINARY SAFETY ASSESSMENT FOR DISPOSAL OF THE DECOMMISSIONING WASTE

Preliminary safety assessments for disposal of the decommissioning waste are conducted. The disposal site considered in the safety assessment is assumed to be located at sea-side of Korea and has stable and homogeneous granite rock formation. Radionuclide inventory of the decommissioning waste discussed in 4.2 are used in the assessment

The safety assessment for the rock cavern-type disposal facility includes groundwater flow evaluation, radionuclide transport and biosphere evaluation. NAMMU program is used in groundwater pathway analysis based on cavern layout. Two-dimensional groundwater flow modelling calculates the most probable pathway between the decommissioning waste disposal cavern and ground water body. MASCOT programme is used to calculate the radionuclide flux released from disposal cavern and the annual individual dose at the geosphere-biosphere interface (GBI). Figure 2 shows the annual individual dose calculated from preliminary safety assessment. Peak dose rate originated from Nb-94 indicates the value of 1.76E-9 Sv/yr at 50 000 yr. In the biosphere modelling, programme BIOS are used to generate dose conversion factors of considered exposure scenario.

The safety assessment for the near-surface disposal facility includes water balance analysis at the engineered cover barrier, groundwater analysis, radionuclide transport analysis and biosphere modelling. Based on the terrestrial climatic condition, HELP programme conducted the water balance analysis of the multi-layered cover barrier. Degradation of soil cover by erosion was considered in balance analysis. System-level safety assessment code, SAGE [4-7], was used to calculate radionuclide transport from the engineered disposal vault to the geosphere-biosphere interface (GBI). In the biosphere modelling, same exposure scenario and dose conversion factors are adjusted in this assessment. Figure 3 shows the annual individual dose calculated from preliminary safety assessment. Peak dose rate originated from Nb-94 indicates the value of 2.47E-9 Sv/yr at 34 000 yr.

Both the rock cavern-type and near surface disposal facility concepts can meet the Korean regulatory dose constraint.



Fig. 2. Annual individual dose calculated from preliminary safety assessment of rock cavern-type disposal facility.



*Fig. 3. Annual individual dose calculated from preliminary safety assessment of near surface disposal facility.*
## 7. CONCLUSION

The decommissioning project for two TRIGA type research reactors in Korea was started in 1997. This paper describes a three-year study on the disposal concepts and waste packaging for the disposal of the domestic decommissioning waste. A survey on the status of decommissioning activities of nuclear facilities has been conducted. Waste packaging and source-term consideration along with decommissioning waste characterization are investigated. Gas generation by corrosion of metals both in the waste itself and in its associated packaging are assessed for all types of waste forms and packages being used. Release of gas from disposal vaults under repository conditions is assessed. Finally, conceptual design and preliminary safety assessment for the proposed disposal facility containing decommissioning waste are also performed.

#### REFERENCES

- [1] MOST Notice 2005-17, "Performance Criteria for LILW Disposal", Ministry Of Science and Technology, Republic of Korea (2005)
- [2] MOST Notice 2005-18, "Waste Acceptance Criteria for LILW Disposal", Ministry Of Science and Technology, Republic of Korea (2005)
- [3] PURDOM, G., Agg, P.J., "GAMMON (Version 1A): A Computer Program Addressing Gas Generation in Radioactive Waste Repositories. Part A: Overview", UK Nirex Ltd. Report NSS/R338 (1993)
- [4] PARK, J.B., et al., "Safety Assessment Codes for the Near-Surface Disposal of Low and Intermediate-Level Radioactive Waste with the Compartment Model: SAGE and VR-KHNP," Waste Management '05, Feb. 27-Mar. 3, Tucson, Arizona (2005)
- [5] PARK, J.W., et al., "Verification of a safety assessment tool for near-surface LILW disposal: System-level test with IAEA Vault Safety Case," International conference on the safety of radioactive waste disposal, Oct. 3-7, Tokyo (2005)
- [6] PARK, J.W, et al., "An Integrated Safety Assessment System for Near-Surface Disposal of Low- and Intermediate-Level Radioactive Waste in KOREA," ICEM '03, The 9th International Conference on Environmental Remediation and Radioactive Waste Management, Examine Schools, Sep. 21-25, Oxford, England (2003)
- [7] PARK, J.B., et al., "Experiences from the Source-Term Analysis of Low and Intermediate Level Radwaste Disposal Facility," Waste Management '03, 23-27 Feb., Tucson, Arizona (2003)

## LITHUANIA

#### DISPOSAL ASPECTS OF LOW AND INTERMEDIATE LEVEL IGNALINA NPP DECOMMISSIONING WASTE

## P. Poskas, A. Brazauskaite, R. Kilda, E. Narkunas, A. Smaizys, R. Zujus

Lithuanian Energy Institute, Nuclear Engineering Laboratory

#### Abstract

The paper presents the main principles, criteria and methods for estimating amounts of contaminated and activated radioactive waste generated during dismantling of technological installations at Ignalina NPP. The improved computer code "DECOM" enabled the recording of the necessary information, performing of the initial data processing and splitting of contaminated waste into different streams based on their dose rates. Rather detailed information about the rooms in controlled area is the bases for performing analysis of the possible waste generation during decommissioning of Ignalina NPP.

Activation modelling of the components of the RBMK-1500 reactor core was performed and preliminary specific activity limits for disposal in planned near surface repository in Lithuania, based on water pathway analysis, have been derived for packages of activated reactor components, such as the shielding and support plates of graphite stack.

1. SCOPE

As part of the coordinated research project "Disposal Aspects of Low and Intermediate Level Decommissioning Waste," the Lithuanian Energy Institute carried out the research project "Disposal Aspects of Low and Intermediate Level Ignalina NPP Decommissioning Waste". Specific areas of the work included:

- Assessment of the characteristics and inventory of the contaminated decommissioning waste streams at Ignalina NPP on a system-by-system basis.
- Calculation of the activation of some reactor components (conservative assumption of impurity quantities)
- Preliminary specification of packages for some activated waste to be disposed of in the planned near surface repository in Lithuania

## 2. INTRODUCTION

There is only one nuclear power plant in Lithuania – Ignalina NPP (INPP). It operates two similar units with a power rating of 1500 MW(e) and a present power level of about 1250 MW(e) each. They were commissioned (first grid connection) in 12/1983 and 08/1987 respectively and provided approximately 70-80% of the electricity produced in Lithuania. The original design lifetime was projected from 2010-2015. On 10 October, 2002 Seimas (Lithuanian Parliament) approved an updated national energy strategy that indicated that the first unit will be shutdown before the year 2005 (following this it was to be shutdown in December 2004) and the second unit in 2009, if funding for decommissioning is available from the EU and other donors. On 26 November, 2002, the Lithuanian government approved an immediate dismantling strategy for Unit 1.

Decommissioning of nuclear power plants is a long and complicated process that requires considerable funds. The preparation for this process also lasts a few years and in case of Ignalina nuclear power plant (Ignalina NPP) means the preparation for safe dismantling of a power plant, the treatment storage and disposal of operational radioactive waste, the storage of spent nuclear fuel, etc. In order to be able to plan the dismantling activities and to introduce radioactive waste processing technologies, storage facilities and repositories, it is necessary to have the preliminary data of the amount of radioactive waste generating during the decommissioning process of the plant, the radioactivity level, nuclide composition and other data.

This work presents the methodology and preliminary results of the assessment of contamination in technological installations by radionuclides, neutron induced radioactivity in some reactor components and the waste streams generating during future dismantlement process at Ignalina NPP. It also presents the preliminary analysis of the possibility of disposing activated reactor metallic components into the near surface repository.

## 3. CONTAMINATED WASTE

During operation of a nuclear power plant not only the reactor itself, but also other systems are being contaminated, such as the main circulation circuit, purification and cooling system, spent nuclear fuel storage pools and others. Their contamination by radioactive particles is due to the circulation of cooling agent (in case of Ignalina NPP – water) in these systems. The water itself is contaminated in the reactor area because of the activation, corrosion processes and defects in fuel cladding. In the case of forced water circulation, radioactive particles in various systems precipitate on the internal walls of system components.

For the assessment of closed systems equipment radioactive contamination, the modified computer code "LLWAA – Decom" (Belgium) was used. The code allows for the determination of the activity  $(Bq/m^2)$  of the deposits located on the system equipment inner surfaces, taking into account the coolant specific activity  $(Bq/m^3)$  and the construction data of system elements (construction materials, geometrical measurements, etc.). It allows also for the calculation of equipment contact dose rate (or the dose rate at the given distance, for example, in case of the presence of thermal insulation). The predicted dose rates can be compared to the measured values at INPP. A good agreement between the predicted and measured equipment dose rates constitutes the basis for the code validation, i.e. of the validation of the predicted deposited activities. Another possibility of validation was the measurement campaign carried out on steel samples removed from the MCC of Unit 1 during the 2002 maintenance outage.

Deposition rate and release rate coefficients are the functions of fluid characteristics (velocity, temperature, Reynolds number), the system equipment characteristics (geometry, inner walls roughness, friction factors), and the characteristics or radioactive particles (its specific weight, diameter). As mentioned above, equipment contamination is concentrated in the surface layer. Contamination occurs due the contact with contaminated coolant. Only the fuel channels (MCC elements) located in the reactor core are contaminated, mostly due to the activation process in the core. The dose rates from different nuclides in reactor water and nuclides deposited on the inner wall of the elements have also been determined. Calculations show that the dose rate from MCC fluid is much smaller than dose rate from deposits. Coolant will be removed before the dismantling process. However, there is an opportunity to compare predicted and measured dose rates even for the systems filled with the coolant.

A detailed assessment of component radiological characteristics was performed for the five most contaminated Unit 1 systems of Ignalina NPP:

The contamination of the remaining system components is rather low. Due to the lack of radiological characterization data, the assessment the radiological characteristics of mentioned components is made conservatively based on existing radioactive measurement data for operational waste, categories of the rooms, etc. (Table 1)

For the assessment and grouping of radioactive waste at Ignalina NPP controlled area the computer code DECOM, developed while preparing the Preliminary decommissioning plan for decommissioning of Ignalina NPP by the efforts of both consortium NIS/SGN/SKB and Lithuanian Energy Institute (LEI), was used. The database of this code includes the data about 42 000 components (or their groups). Later on, this database was permanently complemented and adjusted by more detailed information about the installation data, as well as improving the software by LEI.

Estimated decommissioning waste streams for Ignalina NPP are presented in Figure 1 and Table 1.



Fig. 1. Overall decommissioning waste streams for the whole Ignalina NPP at reactor final shut down. VLLW-SL – very low level waste short lived, LLW-SL – low level waste short lived, ILW-SL – intermediate level waste short lived.

No	Group of components	Waste generated (mass proportion), %			
		VLLW-SL	LLW-SL	ILW-SL	
1	DA (Detailed radiological analysis of systems installations)	24	13.4	62.6	
2	EJM (Engineering judgment of system contamination based on maximum dose rates defined during scheduled maintenance and repair works in that system.	76	10.3	13.7	
3	EJS-1 (Engineering judgment of the contamination of the installations located in the room, based on the maximum dose rate measurements of the operational waste collected in the room ).	97.6	2.4	0	
4	EJS-2 (Engineering judgment of the contamination of the installations located in the room, based on the maximum available contamination assigned to the room category).	91.6	0.3	8.1	
5	Total waste stream mass distribution, %	79.3	6.0	14.7	

# Table 1. Estimated decommissioning waste streams for Ignalina NPP atreactor final shutdown

An analysis of Ignalina NPP decommissioning waste streams shows that it is possible to expect that about 80% of the waste will be VLLW, which could be disposed of into licensed landfill repositories. It is necessary to keep in mind that the waste was split into groups according to the dose rate, which is included in the requirements for radioactive waste treatment at NPP before their disposal [1]. However, this is only a very rough estimation because waste acceptance criteria for landfill repository usually also includes limitations on some of the most important nuclides (especially <sup>137</sup>Cs). Therefore, in the future it will be necessary to assess the nuclide activity of waste and to apply them to the real waste acceptance criteria for the landfill facility. More detailed information on modelling aspects and obtained results is presented in [2, 3]

## 4. ACTIVATED WASTE

As soon as decommissioning work of the reactors of Ignalina NPP begins, the arising radioactive waste will also consist of construction materials of the reactor structure (graphite, concrete, metal parts). These materials are located in and near the reactor core and become radioactive due to the neutron irradiation during the NPP operation. The material composition, neutron flux density and energy distribution in different reactor zones are different, so are the neutron activation conditions. The reactor core (graphite stack, fuel channels tubes, etc.) is the most activated part of the reactor structure, whereas the biological shield (usually concrete and steel structure) is much less activated, as the neutron fluxes are relatively low.

The activity of reactor structure components depends on the initial chemical composition of the particular component material, time and other conditions (e. g. neutron flux) of neutron irradiation and time after final reactor shutdown.

The RBMK-1500 is a graphite-moderated, water-cooled reactor core having a design thermal power capacity of 4800 MW. A schematic section of an RBMK reactor vault is shown in Figure 2.



Fig. 2. Cross-section of the reactor vault [4].

- 1. top cover, removable floor of the central hall
- 2. top metal structure filled with serpentinite 6.
- 3. concrete vault
- 4. sand cylinder

- 5. annular water tank
- 6. graphite stack
- 7. reactor vessel
- 8. bottom metal structure

9. reactor support plates10. steel blocks11. roller supports

The graphite structure consists of 2488 channels, and is made up of columns of bricks each with an axial hole for the channel tube. There are 2052 channels used for fuel, control rods and instruments with the remaining 436 channels around the edge of the core filled with graphite rods to act as the reflector. The entire stack is approximately 8 m high and has a diameter of about 14 m. The four rows of graphite columns at the outer edge make up the radial reflector ( $\sim 1 \text{ m thick}$ ), and a 0.5 m thick layer at the top and bottom make up the end reflectors. Radial displacement of the graphite stack is prevented by 156 water cooled supporting tubes situated at the outer periphery. These tubes are welded to the lower support structure at their base but have the freedom to move in vertical guides at the top. The core is contained above and below by the biological shields and is radially surrounded by a nitrogen/helium gas blanket and water tank. The pressure of the environment is 0.49 kPa in the core cavity and this is lower than outside. Each column of bricks is independent, i.e., there is no keying system. A cup-cone arrangement is used to join and align the bricks end to end. There are steel bricks at the top and bottom of the column. The upper steel bricks are called

shield plates and the bottom steel bricks are called support plates. At the base of the column the bricks are located on a spigot and at the top the column is located in line with a hole in the upper biological shield by means of a telescopic joint. The horizontal joints between bricks are staggered in adjacent channels to avoid any horizontal planes of weakness. The brick has a square cross-section and the central hole has a diameter of 0.114 m. The basic bricks are 0.6 m high although shorter bricks of 0.2, 0.3 or 0.5 m height are also used in parts of the stack. Initially there is very little clearance between the bricks, approximately 1 mm. The fuel cell assembly includes a zirconium pressure tube into which the fuel element assembly is inserted and through which the coolant flows. The pressure tube is located in the central hole of the brick by a system of graphite split rings. Each ring is alternately tight on the pressure tube or tight in the bore of the brick. To prevent oxidation of the graphite and to improve the thermal efficiency, the core is contained in 90% helium, 10 % nitrogen gas mixture. The slots in the graphite rings are aligned to allow the gas mixture to pass along the channel. In this study the model for numerical assessment of Ignalina NPP reactor construction materials neutron activation was developed based on conservative assumptions. Activity inventories were estimated only for graphite stack, channel tubes (to be more precise, for their middle parts, which are made of zirconium-niobium alloy E125), support and shielding plates.

ORIGEN-S computer code (SCALE 5 codes system) was used for the activation analysis [5]. The code considers radioactive disintegration and neutron absorption (capture and fission) and enables to identify isotopic content, activities and concentrations of neutron activated elements.

On the basis of the neutron fluxes measurements carried out in Ignalina NPP Unit 1 reactor core [6] and geometrical dimensions of particular reactor components, it was assumed that thermal neutron flux density remains constant during all irradiation period for all directions in selected component and is equal to:

- ---  $3 \cdot 10^{13}$  n/(cm<sup>2</sup>·s) in the active reactor core (fuel channels, graphite sleeves and blocks in the active core);
- --  $1 \cdot 10^{13}$  n/(cm<sup>2</sup>·s) in the top reflector of the active core (graphite sleeves and blocks in the top reflector);
- $1.5 \cdot 10^{13}$  n/(cm<sup>2</sup>·s) in the bottom reflector of the active core (graphite sleeves and blocks in the bottom reflector);
- $7.5 \cdot 10^{12}$  n/(cm<sup>2</sup>·s) in the radial reflector of the active core (graphite blocks in the radial reflector);
- ---  $6 \cdot 10^{10} \text{ n/(cm}^2 \cdot \text{s})$  in the upper steel bricks (shielding plates);
- -- 9.10<sup>10</sup> n/(cm<sup>2</sup>·s) in the lower steel bricks (support plates).

Figure 3 shows the variation of total specific activities for different reactor components during the 150 year cooling period. It is seen that channel tubes have highest activity concentration during all modelled decay time period. Metal reactor parts have higher specific activities than the graphite parts (bricks and sleeves) at the time of shut down, but after  $\sim 15$  cooling years they reach specific activity levels of active core graphite blocks and sleeves.

The highest activity concentrations for graphite parts are accumulated in active core blocks, sleeves and bottom reflector blocks for all modelled 150 years cooling period.



Fig. 3. The variation of specific activities in different components of INPP Unit1 reactor structure.

Total radionuclide activities in selected reactor components were estimated as the result of the specific activities and mass of each of the components. At the time of final reactor shutdown, the highest overall activity is accumulated in channel tubes. During the first 2 years of cooling the activity of channel tubes decreases significantly, then from this moment to ~ 50 years the decrease is not so high, and for the remaining ~ 100 years the activity concentration stays almost constant. However, the highest overall activity for the modelled time period up to 150 years still is induced in fuel channels. The total activities of metal shielding and support plates are higher than the activity of graphite blocks for the first ~ 10 years of cooling, and reaches the total activity level of graphite sleeves after approximately 30 years of cooling. More detailed information on modelling aspects and obtained results is presented in [7, 8].

## 5. PRELIMINARY SPECIFICATION OF THE PACKAGES FOR ACTIVATED WASTE TO BE DISPOSED OF IN THE NEAR SURFACE REPOSITORY

Preliminary specific activity limits have been derived for packages of activated reactor components, such as the shielding and support plates of graphite stacks. It was assumed that the wastes should be conditioned and disposed of in the near surface repository (NSR) that is planned to be constructed in Lithuania [9]. The derivation of the specific activity limits has been performed using IAEA recommended methodology [10], with respect to requirements of Lithuanian norms concerning radioactive waste management as well as radiation protection.

A waste-leaching scenario with relevant changes of water infiltration rate through the repository during analysed period is considered. Additionally for the task of derivation of activity limits the behaviour of waste form is taken into account, i. e. the container durability of 100 years and further uniform dissolution of activated metallic plates over a period of 100 years (1 % per year) is evaluated [11].

The migration of unit inventory (1 Bq of initial activity for each radionuclide) through the vault and vadose zone has been carried out using the DUST computer code [12] where finite difference method is employed to solve 1-D transport equation with processes of advection,

dispersion and radioactive decay. The assessment of radionuclide transport in aquifer has been performed using GWSCREEN code [13] where 2-D dispersion modelling is implemented.

A potential exposure to local individual of critical group via ingestion of drinking water from well in aquifer has been evaluated. The well is installed at distance of 150 m from the edge of the vault (boundary of the repository). A drinking water consumption of 600 litres per year is supposed.

After modelling of <sup>59</sup>Ni, <sup>63</sup>Ni nuclides migration through the disposal system only impact of the <sup>59</sup>Ni radionuclide has been identified.

Due to rather short half-life (100 years), thick vadose zone (30 m) and high value of distribution coefficient in vadose zone ( $K_d = 0.3 \text{ m}^3/\text{kg}$ ) and in aquifer ( $K_d = 0.4 \text{ m}^3/\text{kg}$ ) the peak concentration value of  $^{63}$ Ni radionuclide in aquifer well (receptor) is less than  $10^{-150}$  orders of magnitude. Therefore the dose induced by  $^{63}$ Ni is negligible. Hence, the preliminary activity limit for  $^{59}$ Ni has been derived.



Fig. 4. Expected doses from radionuclide releases to groundwater pathway.

The preliminary specific activity limit of  $2.08 \cdot 10^5$  Bq/g has been assessed for <sup>59</sup>Ni radionuclide. After comparison of this value to specific concentration values for <sup>59</sup>Ni radionuclide that equal to  $6.45 \cdot 10^2$  Bq/g for shielding plates and  $9.66 \cdot 10^2$  Bq/g for support plates at the time of reactor shutdown, it was concluded that it could be possible to dispose of the activated metallic radioactive wastes into the Lithuanian near surface repository. The expected doses of <sup>59</sup>Ni radionuclide releases from the vault to groundwater pathway are provided in the Figure 4. As the graph shows the maximum value of potential dose is two orders of magnitude lower than dose constraint of 0.2 mSv per year established in Lithuania.

This analysis provides only a very preliminary estimate because, according to Lithuanian regulations [14] and IAEA methodology [10], the activity limits in the case of disposal system evolution (groundwater pathway) as well as the case of inadvertent human intrusion should be evaluated.

#### 6. CONCLUSIONS

- (1) Analysis of Ignalina NPP decommissioning waste streams based on dose rate criteria shows that it is possible to expect that about 80 % of the waste will be VLLW that could be disposed of into licensed landfill repositories.
- (2) The highest overall activity for the modelled time period up to 150 years is induced in fuel channels. Total activities of metal shielding and support plates are higher than the activity of graphite blocks for the first ~ 10 years of cooling and reaches total activity level of graphite sleeves approximately after 30 years of cooling.
- (3) Fuel channel tubes have the highest specific activity during all modelled (150 years) decay time period. Metal reactor parts have higher specific activities than the graphite parts (bricks and sleeves) at the time of shut down, but after ~ 15 cooling years they reach specific activity levels of active core graphite blocks and sleeves.
- (4) Under a conservative estimation of the radionuclide inventory of activated shielding and support plates intended to be disposed of in the vault of the planned near surface repository, the resulting dose to the member of critical group should be two orders of magnitude below the dose constraint defined by Lithuanian norms.

#### REFERENCES

- [1] Requirements for radioactive waste treatment at nuclear power plant before their disposal. VD-RA-01-2001. VATESI, order No. 38, 2001.07.27.
- [2] POŠKAS, P., ZUJUS, R., BRAZAUSKAITĖ, A., Preliminary radiological characterization of the main circulation circuit at Ignalina NPP for decommissioning purposes, Proc. of International Conf. "Waste Management 2004", USA, Tucson, Arizona, 2004.
- [3] POŠKAS, P., ZUJUS, R., KOLESNIKOVAS, J., Principles for estimating the radioactive waste from decommissioning at the Ignalina nuclear power plant, International Symposium "Disposal of Low Activity Radioactive waste", Cordoba, Spain, December 13-17, 2004.
- [4] ALMENAS, K., KALIATKA, A., UŠPURAS, E., Ignalina RBMK-1500. A Source Book, Kaunas: LEI, 1998.
- [5] GAULD, I.C., HERMANN, O.W., WESTFALL, R.M., ORIGEN-S: SCALE system module to calculate fuel depletion, actinide transmutation, fission product buildup and decay, and associated radiation source terms. Version 5, Oak Ridge National Laboratory, April 2005.
- [6] Ignalina NPP Safety Assessment Report.-Ignalina, 1996.
- [7] NARKŪNAS, E., ŠMAIŽYS, A., POŠKAS, P., Numerical modelling of neutron activation processes in the reactor RBMK-1500 structural elements: 1. Numerical modelling of neutron activation processes in fuel channels (alloy E125), Energetika – 2005, No. 1. (in Lithuanian).
- [8] SMAIZYS, A., NARKUNAS, E., POSKAS, P., Modelling of activation processes for the GR-280 graphite at Ignalina NPP, Radiation Protection Dosimetry (accepted for publication).
- [9] Environmental impact assessment report for construction of a near surface repository for radioactive waste. LEI, GGI, Vilnius, 2005. (in Lithuanian).
- [10] INTERNATIONAL ATOMIC ENERGY AGENCY, Derivation of Activity Limits for the Disposal of Radioactive Waste in Near Surface Disposal Facilities, IAEA-TECDOC-1380, IAEA, Vienna (2003).

- [11] SULLIVAN, T.M., Challenges in source term modelling of decontamination and decommissioning wastes // Presentation at IAEA workshop "CRP on disposal aspects of decontamination and decommissioning wastes", September 13-17, 2004.
- [12] SULLIVAN, T.M., Disposal Unit Source Term (DUST). Data Input Guide. US Nuclear Regulatory Commission Report NUREG/CR-6041, Brookhaven National Laboratory Report BNL-NUREG-52375. Brookhaven National Laboratory, Upton, New York 11973, 1993.
- [13] ROOD, A.S., GWSCREEN: A semi-analytical model for assessment of groundwater pathway from surface or buried contamination. Theory and user's manual, Version 2.0. EGG-GEO-10797, Revision 2. Idaho National Engineering Laboratory, 1994.
- [14] Generic Waste Acceptance Criteria for a Near Surface Repository. VATESI, P-2003-01, State Journal, 2003, No. 19-850.

#### **RUSSIAN FEDERATION**

#### RESEARCH PROJECT OF SAINT PETERSBURG INSTITUTE OF TECHNOLOGY: THE MAIN RESULTS

#### A.F. Nechaev, Yu.S. Koryakovskiy

St. Petersburg State Institute of Technology (Technical University), Department of Engineering Radioecology and Radiochemical Technology

#### Abstract

St. Petersburg State Institute of Technology, being one of the Rosatom research and educational centres, for many years has actively worked in various areas of radioactive waste management, including (1) development of strategic approach to decommissioning waste management, (2) assessment of inventory of decommissioning waste streams, (3) development of advanced technologies for predisposal treatment and conditioning of decommissioning waste, (4) investigations of barrier properties of materials for isolation of decommissioning waste, and (5) development of methodology for safety insurance and performance of radwaste repositories. An opportunity to obtain additional knowledge on the subject, on one hand, and to share the experience gained with the colleagues, on the other hand, were the stimulus for the research group to join the IAEA CRP. Research Project "Integrated Approach to Decommissioning Waste Life-Cycle: Methodology and Technology" has been conducted in full compliance with the CRPs Terms of References, with the ultimate goal "to promote transfer of knowledge, relating to the data that are important in the planning for the disposal of decommissioning waste".

#### 1. SCOPE OF THE PROJECT

## Key targets

During the mass decommissioning of nuclear facilities, disposal of the waste generated should be considered as an issue of high priority. In other case, liquidation of one nuclear facility (e.g. NPP) will lead to creation of another one (e.g. radwaste storage facility) with indefinite future. This obvious circumstance has predetermined both the main aims and algorithm of present investigation. Research Project entitled "Integrated Approach to Decommissioning Waste Life-Cycle: Methodology and Technology" has addressed the following objectives:

- to estimate the scope of decommissioning activities in Russia and inventories of decommissioning waste supposed to be generated in the course of these activities;
- to analyze characteristics and peculiarities of decommissioning waste, and to define uncertainties to be solved;
- to investigate feasibility of various options for the disposal of decommissioning waste, to determine important issues impacting or potentially able to impact requirements for and design of waste repositories;
- --- to create criteria/indicators and corresponding database in support of proper source term modelling and safety assessment of decommissioning waste disposal.

# 2. SCOPE OF DECOMMISSIONING ACTIVITIES IN RUSSIA

The scope of decommissioning activities in Russia is enormous. Just ongoing projects involve:

- 190 nuclear submarines (altogether about 380 reactors) and a few scores of auxiliary vessels for nuclear-technological service (mostly for temporary storage of radwaste and spent nuclear fuel);
- four nuclear power units (already shutdown and transferred in care-and-maintenance regime);
- 27 research reactors and critical assembles (shutdown, defueled, and dismantled or transferred into safestore state);
- 20 radwaste storage open ponds with total capacity of a few hundred millions cubic meters (in various stages of decommissioning/liquidation);
- one of sixteen radwaste management enterprises of "Radon" system;
- --- ten from 13 plutonium-production reactors (shutdown and transferred in care-and-maintenance regime);
- 130 units of radioisotope thermoelectric generators;
- a few scores of research and industrial radiation facilities (under dismantling and demolishing).

The need for decommissioning nuclear facilities will increase over time. For instance, towards 2020 practically all currently operated nuclear power plants will be formally subject to decommissioning (Table 1); and the fate of aging "Radon" enterprises, located in (or very close to) highly populated areas, will require serious attention. The time is approaching plan for the future of the large nuclear fuel cycle plants.

NIDDa nomo	Number	Type of	Power,	Commission	Planned date of
INF F S Hallie	of Units	reactor	MW	Commission	decommissioning
Belojarskaja	3	BN-600	600	1980	2010
Bilibinskaja	1		12	1974	2004
	2	EGP-6	12	1974	2004
	3		12	1975	2005
	4		12	1976	2006
	1		1000	1985	2015
Dalakovskaja	2	WWER-1000	1000	1987	2017
Dalakovskaja	3		1000	1988	2018
	4		1000	1993	2023
	1	WWER-1000	1000	1984	2014
Kalininskaja	2		1000	1986	2016
	3		1000	2004	2034
	1		440	1973	2003-2008*
Kol'skojo	2	WWER-440	440	1974	2004-2009*
KUI SKAJA	3		440	1981	2011
	4		440	1984	2014
	1		1000	1976	2006-2011*
Vurskojo	2	RBMK-1000	1000	1979	2009-2014*
Kuiskaja	3		1000	1983	2013
	4		1000	1985	2015
	1		1000	1973	2003-2008*
T	2	RBMK-1000	1000	1975	2000-2010*
Lennigrauskaja	3		1000	1979	2009-2014*
	4		1000	1981	2011
Novovoronozhekoj	3	WWER-440	417	1971	2001-2005*
novovoroneznskaj	4		417	1972	2002-2007*
a	5	WWER-1000	1000	1980	2010
	1		1000	1982	2012
Smolenskaja	2	RBMK-1000	1000	1985	2015
	3		1000	1990	2020
Rostovskaja	1	WWER-1000	1000	2001	2031

Table 1. Provisional schedule of NPPS decommissioning

\*) In provision of plant life extension

It must be understood that the decommissioning activity on such a scale will lead to the generation of enormous quantities of radioactive waste. Proper management of this waste is going to be one of the key issues of decommissioning activities. As it was correctly stated in [1] "it is impossible to discuss decommissioning without reference to waste disposal".

# 3. ESTIMATION OF DECOMMISSIONING WASTE ARISINGS

# 3.1. Volumes of decommissioning waste

According to [2-4] it is expected that in the course of decommissioning activities in the next 20 years, there will be generated around 1.6-2.5 mln. tons of radioactive (or suspected to be radioactive) metals; 30 000 tons of graphite, 120 000-150 000 m<sup>3</sup> of decontamination liquid

waste, millions tons of concrete requiring at least careful radiological examination, a large volumes of contaminated cables, heat insulation, electric insulators, plastics, etc.

Unfortunately, analysis of information on decommissioning waste arisings clearly indicates serious discrepancies in the estimates provided by various authors. For example, estimated volumes of compacted decommissioning L/ILW from the reactors of the same type (WWER-440), in terms of deferred decommissioning approach, varied from 2000 to 14 000 m<sup>3</sup>; quantities of decommissioning solid and solidified waste from RBMK-1000 reactor – from 18 000 to 100 000 m<sup>3</sup>; decommissioning of uranium enrichment plant is expected to produce from 12 700 m<sup>3</sup> to 230 000 tons of radioactive waste.

The main reason for such uncertainties is the absence of a unified methodology for the evaluation of decommissioning waste arisings, and this issue deserves very serious attention.

In the context of the present study, it was reasonable to confine ourselves to a semiquantitative comparison of operational and decommissioning radwaste volumes. The results of such a comparison inevitably lead to the following conclusion: the **quantity of decommissioning waste to be disposed can not be less than the total volume/mass of operational conditioned waste accumulated during the life-time of facility, but in many cases one can expect an excess of "initial" volume of waste with factors varying from several to several tens of times.** For instance (see Fig. 3.1):

- (1) The total volume of solid operational waste accumulated at 31 Russian power reactors comes to 110 000 m<sup>3</sup>. It is planned (and possible!) to reduce this volume as much as 4-5 times [5]. At the same time, the most "optimistic" estimation of decommissioning solid waste arising results in an amount of about 20 000 m<sup>3</sup> per unit and, consequently 620 000 m<sup>3</sup>, in all.
- (2) The mass of operational solid waste stored at the naval technical bases has reached 20 000 tons, while the total weight of radioactive structural materials, expected to be generated during the dismantling of nuclear submarines, is estimated in 400 000 tons [6]. For long-term storage and, possibly, for further disposal of reactor compartments of decommissioned submarines, it is necessary to have disposal capacity of about 390 000 m<sup>3</sup> [7].
- (3) Dismantling of the Russian research reactor WWR-S resulted in the generation of 620 tons of decommissioning waste, while the total mass of operational waste did not exceed 14 tons.
- (4) The approximate mass of radioactive operational waste from all French NPPs is estimated to be as much as 36 times less than the total mass of expected decommissioning waste [8].



Fig. 1. Ratio of the waste volumes: decommissioning/operational:

- 1. Russian nuclear submarines (without waste that suspected to be radioactive),
- 2. Russian NPPs (after the planned reduction of operational waste volumes),
- 3. French NPPs (prognosis),
- 4. Research reactor WWR-S (real data).

## 3.2. Potential impact of operational waste

An essentially larger volume of decommissioning waste in comparison with operational waste is not the only peculiarity of decommissioning waste. During decommissioning of radwaste storage facilities, the waste in storage is categorized as *decommissioning waste* (by definition). This is typical for decommissioning of "Radon" companies, waste storage tanks and open ponds at radiochemical plants, and waste storage damps at uranium enterprises. It is also applicable to any other nuclear facilities if *during the period of "safe enclosure" operational wastes are stored in reactor building* (NPP or plutonium production reactors) or in *reactor compartments* (nuclear submarines) as it is planned in Russia [9-11]. Thus, *operational waste can contribute essentially to volumes of decommissioning waste*.

# 3.3. Potential impact of "Clean" and "Clearable" Waste and Materials

It is generally assumed that the bulk the waste and materials generated during dismantling and demolishing operations are dominated by non-radioactive waste and materials that could *potentially* be released for unrestricted or conditional use. According to some estimates [12] only 2% of the total "decommissioning" mass from NPP is attributed to radioactive waste, 6% is attributed to the materials clearable after decontamination, and 92% is attributed to clear materials. If so, this would radically influence the demand for disposal capacities and decrease disposal costs as much as 5-15 times. However *expectations of dramatic waste minimization at the expense of careful segregation and/or decontamination of material are still debatable*. In any case, there is still much work to be done towards further developments.

Specifically, recycle and reuse of metal from the nuclear sector depends on traceability, legitimacy and public perception. These preconditions, in turn, require acceptable and enforceable criteria for the release of low-activity contaminated materials, i.e. *internationally* agreed criteria that are protective of health and safety while providing flexibility within the spectrum of recycling and reuse of materials that pose little or no hazard to the general public.

Previous attempts to establish coordinated dose limits for unrestricted release of materials associated with - but not necessarily contaminated by - nuclear facilities and operations *have met with failure* [13], and now the main task is to seek consensus, because the illicit trafficking of contaminated metals presents transboundary concerns.

It must also be remembered that difficulties have been encountered in obtaining acceptance of cleared material by scrap dealers or commercial smelters, who refuse this kind of raw material for their production. The combination of these factors – *absence of internationally agreed upon criteria, a tense attitude in the metal industry and in public media* – should not be ignored when considering the prospects for the recycle of metal from decommissioning activities.

Another capacious stream of decommissioning residues is concrete, the greater part of which, as a rule, is not contaminated or activated at all, or can be decontaminated relatively easily. However, opportunities for reuse of "decommissioning" concrete are, apparently, overestimated to a considerable extent, because (1) the possibilities of utilizing concrete rubble are rather limited, and (2) it is unlikely that "decommissioned" concrete will be recycled by the construction industry. Thus, there is a rather high probability that a substantial share of "decommissioning" concrete must be disposed of, which requires appropriate attention.

The difficulties facing recycle and reuse of materials generated during dismantling of nuclear facilities could lead to a sharp rise in the volume of decommissioning waste to be disposed. The most obvious consequence of this is the re-classification of cleared materials as very low level waste (VLLW). However, VLLW is not a formally recognized category either in the European Union (excluding France) or in Russia. Therefore, there is no defined disposal option for this material/waste other than as low level waste. In practical terms it means that *demands for LLW disposal capacities could increase significantly*.

Disposal of high volumes of "clean" decommissioning waste also requires careful consideration, because:

- (1) It may require construction of new landfills with all that it implies: site selection, environmental safety assessment, concordance, licensing, etc.
- (2) Clearance for disposal of decommissioning and conditional industrial waste may differ what, in turn, may give rise to serious problems of regulatory and organizational character.
- 3.4. The rate of generation and heterogeneity of decommissioning waste

The bulk of decommissioning waste is generated during the stage of dismantling and demolishing of a nuclear facility. This stage, once initiated, can not be prolonged over a long period of time *because dismantling, in essence, is the process of the consecutive destruction of physical barriers of safety.* Therefore, a predominant share of decommissioning waste is expected to be produced within a period of one to five or seven years, depending on the size and constructive peculiarities of the facility. It is also important to note that decommissioning waste, produced in dismantling operations almost simultaneously, exhibits substantial differences in radiological characterization, size, density, physical conditions, toxicity, chemical properties, etc. The main challenge presented by decommissioning is the generation of substances/materials *practically at the same time, and in large volumes.* Because of this, it would be logical for decommissioning waste management to give preference to the "pack and removal" option over the "treat and storage" option. However, this approach requires an

availability of repositories or, at least, clarity with respect to disposal routes for waste including non-radioactive residues as well as hazardous and toxic waste. Otherwise, dismantling of one nuclear facility can lead to the creation of other nuclear and/or hazardous material(s) in the indefinite future.

# 4. WASTE DISPOSAL OPPORTUNITIES AND REQUIREMENTS

## 4.1. On-going projects and planned activity

Recent estimates (2004) in Russia indicate the following accumulation of waste: more than  $4.15 \cdot 10^8 \text{ m}^3$  of liquid radwaste, around  $7.3 \cdot 10^8$  tons of solid and solidified waste, and 14 600 tons of spent nuclear fuel [14]. The absolutely predominant share of these wastes is concentrated at enterprises of the Federal Atomic Energy Agency. Liquid radwaste of the Federal Atomic Energy Agency (FAEA) is distributed over 98 facilities at 32 enterprises; solid waste is distributed over in 273 facilities/sites at 39 enterprises. In addition, there are 3 repositories for deep underground storage of liquid radioactive waste with a total capacity of about 50 mln. m<sup>3</sup> [15], and these facilities could be considered as the only repositories in operation.

Summing up the above discussion, one could conclude that presently the disposal activity in Russia is limited by the acute shortage (or better to say – absence) of repositories, even for operational radioactive waste.

In the last few years, projects on site selection (for) and design of the regional and/or centralized (interregional) repositories for both operational and decommissioning waste have been initiated in a number of regions of Russia: from Far-East to the North-West of the country. The most intensive programmes for disposal of L/ILW (both operational and decommissioning) are realized in the North-West region of Russia. A comparative analysis of potential radwaste disposal sites has been carried out for three types of geological formations [16-19]:

- (1) Precambrian metamorphic thick series and large scale granite intrusions (the Murmansk area territory, the Cola Peninsula);
- (2) Palaeozoic folded complexes of the Novaya Zemlya archipelago within the permafrost development area;
- (3) Phanerozoic sedimentary cover rocks of the Russian Platform (the Leningrad area).

An assessment has been carried out of the long-term stability of the rock masses and possible changes of the geological situation for a period of many hundreds years. Both external (i.e., geological, climatic) and internal (i.e., radiation) factors, which are specific to the underground repository, have been taken into account.

In addition to the active investigations aimed at proper site selection, a number of design solutions have been developed for near-surface radwaste repositories, in principle, for disposal of low and intermediate level decommissioning waste. In this context one could mention:

Moscow SIA "Radon" simple storage facility (Fig. 4.1) and "dry-wells" of a large diameter that, under certain conditions, can be converted in the near-surface repositories (Fig. 4.2);

- project of VNIPIET two versions of repository based on Saint-Petersburg metro (underground) technologies (Fig. 4.3);
- original vertical facility initially intended for temporary storage but with some prospects to be transformed in the near-surface repository [20], etc.



Fig. 2. Advanced repository for conditioned waste.



Fig. 3. Disposal of radwaste in big-diameter boreholes.



Fig. 4. The element of underground radwaste repository.

# 4.2. Specific options for decommissioning waste disposal: practical experience and prospects

So called "specific options" include *in-situ and/or on-site disposal* of decommissioned waste (when there is no necessity to construct and operate special repositories), as well as *deferred disposal* of very large components (like reactor compartments of nuclear submarine). These options, as a rule, can be applicable in the case where the nuclear facility is situated far from the populated localities, and in an area with suitable geological and hydrological conditions.

Reconstruction of some research reactors presents the most visible examples of in-situ disposal of decommissioning waste. For instance, during decommissioning of uraniumgraphite reactor RPT it was decided to construct a new research reactor MR *in the same building*, and after the dismantling of technological systems, *the frame of RPT reactor together with the graphite brickwork was grouted with concrete "in-situ"*. The same approach (in-situ disposal) is used in relation to open radwaste storage ponds located at the IA "Mayak", Siberian Chemical Integrated Plant, and Mining-Chemical Integrated Plant [21]. The in-situ disposal approach is also planned to be employed for *industrial (plutonium production) reactors and reactor compartments of nuclear submarines* [11].

The on-site disposal option, to some extent, is similar to the in-situ method, and it was also used in real practice. For instance, in 1984 it was decided to close site "Galit" (salt dome of Bolshoi Azgir), which had been used for testing of nuclear-explosive technologies to create caverns of large capacity in the massif of the rock-salt. Incidentally, the requirement was to restore the testing site to the "green field" state of unrestricted use. To dispose of radioactive waste generated in the course of decommissioning operations, the cavern A-X was used, which had a 100 000 m<sup>3</sup> capacity and was created by the detonation of a nuclear appliance at a depth of 982 m [22]. The volume of radioactive waste accounted for was about 24 300 m<sup>3</sup>, with a total activity ~2000 GBq.

The same approach (on-site disposal of decommissioning waste in the man-made caverns or bore-holes) is under development in the oil and gas industry, and in particular at production sites located in zones of the past underground nuclear explosions. At present this project is in the final stage of development in the "Lukoil-Permoil" company [23], as an alternative to the previously employed technology: decontamination of equipment and disposal of the radwaste generated as well as contaminated soil in the shallow repository of ~ 6000m<sup>3</sup> capacity [24, 25].

On-site disposal approach is also intended to be used for some reactor facilities. For instance, the 100 kW pool-type research reactor RG-1M was shutdown, and its reactor fuel was unloaded and transported to radiochemical plant "Majak" in 1999. After dismantling, in-reactor equipment and structures will be transferred to the radwaste storage / disposal enterprise "Radon". Operational and decommissioning low level radioactive waste are intended to be disposed of *onsite* – in an existing radwaste storage facility (an underground reinforced concrete compartment 6x9 m in area and 5.5 m deep with a 5.5.mm thick steel liner and one upper inlet) and in the shaft with the reactor pool inside, from which all the equipment has been removed and the coolant drained. Both facilities will be properly isolated (covered by 500 mm concrete "pillow" with hydro isolation and by 1.5-2.0 m of the bedrock to prevent the thawing out of the permafrost soil in summer period). Radiological monitoring and later – institutional control of the near-surfaces repositories will be provided [26].

With regard to decommissioning of nuclear submarines – the most topical and currently the largest-scale problem – the final solution for disposal of reactor compartment (*11 m in diameter and 11-12 m in length*) is not yet accepted. This case is noteworthy because it concerns one of the key questions of decommissioning waste disposal: how to manage large parts (i.e. reactor vessels) of nuclear facilities? A temporary, *but long-term*, solution is to store reactor compartments at the specially constructed open sites. The storage period is planned to be 70-100 years. Reactor compartments are placed on special supports. The storage site is equipped with moorage, self-propelled transportation system, system of water supply, sewage system, a special system of collection and isolation of corrosion products, communication system, fire-alarm system, technical means of physical protection and security, and a system for radiation control of ground water around the site. It is important that reactor compartments will be used also as *storage facilities for solid decommissioning waste*. In each compartment it is foreseen to load 40-100 m<sup>3</sup> of solid radwaste with a total activity up to 8 10<sup>11</sup> Bq.

This example once again shows that disposal of radioactive waste (and decommissioning waste – in particular) is not a simple task, but requires careful consideration of a multitude of various factors.

## 5. PREDISPOSAL TREATMENT OF LIQUID DECOMMISSIONING WASTE

The quantities of expected decommissioning waste are enormous, and it is important to undertake all possible measures to minimize the volume of waste to be disposed. During decommissioning activities, the main source of liquid radwaste is spent decontamination solutions. As a rule, such fluids contain *complexing organic constituents* that make it practically impossible to employ such traditional techniques as adsorption or ion-exchange for isolation of polyvalent radionuclides.

Thus, removal of complexing organic constituents is one of the key preconditions for the deep treatment of liquid radwaste, and by this way – effective volume reduction. Technology described below allows for the *reduction of the volume of decontamination fluids by at least 100-200 times*.

The key component of the treatment facility is the apparatus of electro-stimulated destruction of organics. The original idea of this facility is based on the following simple reactions of cathodic reduction of molecular oxygen:

$$\begin{cases} O_2 + 4H^+ + 4e^- \rightarrow 2H_2O \\ O_2 + H_2O + 4e^- \rightarrow 4OH^- \\ O_2 + 2H^+ + 2e^- \rightarrow H_2O_2 \\ O_2 + H_2O + 2e^- \rightarrow HO_2^- + OH \end{cases}$$

The cathode is fabricated from a special material, the anode from the lead dioxide or graphite; oxygen is supplied in the system by pumping air. The main advantage is the intrinsic safety of such "destructors" with respect to water electrolysis with generation of hydrogen.

Simultaneously with destruction of organic constituents, radionuclides are precipitated along with a selected co-precipitator. Depending on the initial composition of radioactive liquids, an additional ion-exchange column for final purification may be required. Thus, the proposed facility produces non-radioactive organic-free solution containing dissolved ions. These remaining constituents may represent a chemical, but not radiological, hazard.

The "electro-stimulated destruction" (ESD) method is compatible with ozonation effectiveness, and it also enables the user to reduce energy consumption 6-20 times (depending on the nature of compounds destroyed), to decrease the dimensions of the facility 40 times, to cut initial investments and operational expenses; it does not require utilization of pure oxygen, it can be applied for the solutions with the salts content, at least, up to 400 g·dm<sup>-3</sup>, and it guarantees a principal impossibility of the "ratting mixture" generation.

Pilot-scale trials have clearly demonstrated that employment of the ESD method for processing of radioactive liquid concentrates, and in some cases, allows the user to completely exclude the ion-exchange decontamination stage, since polyvalent radionuclides are separated by co-precipitation and subsequent filtration [27-29]. Electricity consumption can be assessed as around 20-25 (A-hour) dm<sup>-3</sup>.

The technology and techniques for deep treatment of liquid decommissioning radwaste has been employed in a special modular facility developed in the framework of St. Petersburg R&D Initiative. For more detailed information see [27-32].

6. POTENTIAL COMPLEMENTARY INDICATORS FOR SAFETY ASSESSMENT OF DECOMMISSIONING WASTE DISPOSAL

Presently, great varieties of materials for siting/construction of radwaste repositories are investigated and employed or planned to be employed in practice. The unique issues associated with radioactive waste management (in comparison, for example, with chemical waste management) is that requirements for materials used in storage/disposal of radwaste do not have a prescriptive character, and the applicability of certain engineering barrier or hosting rock is defined only by performance indicators.

In general, to select the proper material and to evaluate the reliability of this material during the required period of time it is necessary to know the "intrinsic permeability" of material for a certain radionuclide or group of radionuclides expressed in some universal units, characterizing mobility (migration rate) of nuclides in a given environment.

Presently, barrier properties of materials and hosting environment of repositories are evaluated on the basis of the leaching rate, adsorptive properties, and sometimes diffusion coefficients measured, as a rule, at high temperatures. Incidentally, as we know, there are no comprehensive catalogues or reference books on the comparative "permeability" of commercially available or innovative materials for, at least, so called critical radionuclides. At the same time, there is an obvious necessity to develop proper indicators for the comparative assessment of the barrier properties of materials used in radwaste management practice. For this purpose it was proposed:

- (1) To collect, systemize and analyze information on enthalpies and coefficients of diffusion of various radionuclides (or stable isotopes) in a broad spectrum of artificial and natural materials (metals, alloys, glasses, cements, various ceramics, Synrock-like materials, clays, granites, sandstones, limestones, bentonites, etc.).
- (2) To extrapolate all the data (both found in the literature and obtained in the laboratory) to the fixed temperature for instance to  $20^{\circ}$ C.
- (3) To calculate on the basis of the data collected the time required for migration of the definite share of radionuclides on the fixed distance at the constant temperature.
- (4) To verify the applicability of this criterion.
- (5) To organize all the data in such a way as to provide the parties concerned with the opportunity to carry out preliminary selection of the barrier material, to extrapolate the data for selected materials to the real temperatures, and to calculate the rate of radionuclide(s) migration in order to assess the necessary thickness of the isolating wall or any other parameters required for engineering analysis.
- (6) To prepare a reference book on the barrier properties of materials used in radwaste management practice.

Table 2 contains certain data demonstrated variability of *proposed complementary safety indictors* ( $T_1$  – time required for migration of the definite share of radionuclides on the fixed distance at constant temperature, and  $L_1$  – the distance of radionuclide migration during the set time at constant temperature) for various materials/media and various radionuclides

Material/media	Diffusant	$T_{l}$ , years <sup>*</sup>	$L_l, cm^{**}$
	<sup>85</sup> Sr	3.1-3.8	89.4-98.6
Compressed Na-bentonite saturated with	<sup>137</sup> Cs	29.3-48	25-32
natural ground water (10 g/L)	<sup>60</sup> Co	418-1465	4.5-8.5
	<sup>125</sup> I	122-266	10.6-15.7
	<sup>85</sup> Sr	2.4	110.9
	$^{134}Cs$	14.7	45.3
	<sup>232</sup> Pa	48.8	24.8
Compressed Na-bentonite saturated with	<sup>233</sup> U	35.7	29
synthetic water	<sup>234</sup> Th	3447	3
	<sup>237</sup> Np	33.7	29.8
	<sup>239</sup> Pu	976	5.5
	<sup>241</sup> Am	2093	3.8
	<sup>95m</sup> Tc	0.9	181
	<sup>225</sup> Ra	5.2	75.7
Bottom sediment from the Atlantic	<sup>228</sup> Th	2930	3.0
Ocean (70-90% of clay), saturated with	<sup>237</sup> Np	8.9	58.1
sea water	<sup>239</sup> Pu	1628	4.0
	<sup>241</sup> Am	366	9.0
	<sup>243</sup> Cm	2930	3.0
Granite saturated with synthetic water,			
from:			
	137 ~		
1. Finnsjon	<sup>137</sup> Cs	225.0	11.0
	241	37.0	28.0
	<sup>241</sup> Am	10460-12740	1.5-1.7
2. Studsvík	137Cs	172	13.0
3. Stripa	$^{137}Cs$	553.0	7.0
	<sup>125</sup> I	349.0	9.0
	<sup>241</sup> Am	7710	2.0
Portland cement with the			
"cement/water" ratio:			
0.2	1370-	2020	2.0
0.2	$\frac{Cs}{^{137}Ca}$	2930	3.0
0.43	$\frac{Cs}{137}$	29.5	32
Highly porous concrete		0.5	520
1. with open surface	<sup>222</sup> Rn	0.00004	26390
2. with 3 layers of latex paint	<sup>222</sup> Rn	0.0002	11100
3. with 3 layers of synthetic paint	<sup>222</sup> Rn	0.003	3050

# Table 2. Variability of complementary safety/performance indicators in some materials

\*In this example, distance is 10 cm, and temperature is  $20^{\circ}$ C.

<sup>\*\*</sup>Settled time is 300 years, temperature is  $20^{\circ}$ C.

In total, the database contains a few thousand units of carefully processed information. These materials (in Russian) are available upon request. In 2006 it is planned to produce an English version of the database, and to present it on the Internet.

# **KEY FINDINGS**

During implementation of the Project, two major missions have been carried out simultaneously and with close interdependence:

- gathering and detailed analysis of relevant information (info-analytical route),
- Research and Development (R&D) activities.

The main results of the info-analytical route are following:

- Performing of analysis of the overall situation in the country, which led to an assessment of the level of readiness for the large-scale decommissioning programmes and a definition of the real needs, opportunities and problems associated with decommissioning waste arisings and disposal;
- Development of a conceptual approach for the preparation of decommissioning waste for disposal supported by relevant databases;
- Assessing the feasibility of various options for decommissioning radwaste disposal;
- Collateral participation in the decision-making process through advice of authoritative bodies, expert appraisals, improvements of educational and training programmes.

Research and developments (R&D) have allowed:

- for the development of an advanced predisposal treatment and conditioning technology (for some types of decommissioning waste) compatible with repository conditions;
- for the performance of an investigation of barrier properties of various materials applicable to the conditioning /solidification/containerization of decommissioning liquid; for solid waste, to provide the scientific basis for the proper selection of materials and host rocks; and to allow for an improvement in the quality of prediction of decommissioning waste package performance over time and the safety of the disposal system, in general.

Work on analysis and assessment had led to an important set of conclusions. First of all, the volumes of decommissioning waste liable to disposal, as a rule, essentially exceed the quantities of operational waste accumulated during the entire life-time of the facility. More importantly, under certain conditions operational waste can be incorporated with decommissioning waste. In addition, provisionally "pure" and decontaminated materials may seriously contribute to these volumes if regulatory and some technical issues of recycling and reuse are not solved in a reasonable and timely manner.

These facts must be taken into account at the time of determining disposal capacities required. It is also important to foresee in advance disposal routes for non-radioactive decommissioning waste (including chemically toxic materials) that are generated in large volumes as well.

Important special feature of decommissioning process is the high rate of the waste generated, which stimulates an employment of the "pack and remove" approach to waste management. In order to prevent an extremely undesirable situation in which one nuclear facility (e.g. NPP) is replaced by another (off-site storage facility) in the indefinite future, *at the time of dismantling* it is necessary to have a repository or, at least, full clarity with respect to disposal routes for all kinds of decommissioning waste.

It is also very important to recognize the principal heterogeneity of the waste generated at the stage of dismantling. All peculiarities associated with the diversity of properties, sizes and forms of the waste have to be taken into account in the siting and design of repository.

## 7. CONCLUSION

Implementation of Research Project 12184 "Integrated Approach to Decommissioning Waste Life-Cycle: Methodology and Technology" in the framework of the IAEA CRP "Disposal Aspects of Low and Intermediate Level Decommissioning Waste" has resulted in the development of a set of analytical and technical materials that are important for timely and well-considered planning and preparation of a large-scale decommissioning programmes in Russia. Some aspects of decommissioning waste disposal revealed and discussed in this IAEA supported study, have not even been taken into consideration up until now. Presently, these issues became the subject of discussion at representative scientific and technical forums (e.g. Annual International Conference on Safety of Nuclear Technologies; International Science and Technical Conference on Radwaste Management, etc.), and in responsible organizations.

Thus, participation of the Saint-Petersburg State Institute of Technology's research group in joint investigation with, initiated and co-ordinated by the IAEA, have allowed for the stimulation of "internal" activity in the practically important sector of nuclear technology – safe decommissioning waste disposal. Apparently, this is the most visible and the most significant result of Research Project that has been successfully performed owing to the initiative and the aid of the International Atomic Energy Agency.

## REFERENCES

- [1] CANADIAN ASSOCIATION OF NUCLEAR HOST COMMUNITY, Larry Kraemer, Chairperson of CANHC and Mayor of Municipality of Kincardine, 2003. Adobe file: Kraemerwithnotes.
- [2] PASTUSHKOV V.G., et al., Technology and Equipment for Processing of Radioactive Metals in Cold Crucible. Proc. 4-th Intern. Conf. on Radioactive Waste Management (26-28 June 2001, Moscow). M.: VNIIAES, 2002, p. 99-107.
- [3] SIMANOVSKY V.M., Integrated Engineering and Radiation Surveillance of Nuclear Facilities Intended for Decommissioning. Ecology and Atomic Power, 2001, N 1, p. 72-73.
- [4] TSYKOV S.F., Safety Problems in Industrial Waste Management at FSUP "Sevmash". Proc. Intern. Seminar "Ecological Problems of Nuclear Submarines Decommissioning" (4-9 June 2001, Severodvinsk). SD, 2001, p.61-66.
- [5] BYLKIN, B.K., CHAPOSHNIKOV, V.A., KLUTCHNIKOV, B.V., TISHKOV, V.M., Operational Radioactive Waste Management in Leningrad NPP. Ecology and Atomic Power, (2004), №1 (4), p.40-45.
- [6] DOVGUSHA, V.V., TICHONOV, M.N., On Potential Nuclear and Radiation Hazard in the Course of Utilization of Nuclear Submarines and Ships Equipped with Nuclear Power Units. Proc. 5-th Intern. Conf. "Radiation safety" (24-27 September 2002, St.Petersburg). SPb: Restec (2002), p.226-242.
- [7] KULIKOV, I.D., et al., Radiation Safety Assurance in Decommissioning of Reactor Units under the "Disposal" Option. Proc. 3-B Intern. Conf. "Radiation Safety" (Oct. 31-Nov. 4, 2000, St.Petersburg). SPb: Restec (2000), p.238-241.

- [8] DUTZER, M., The Decommissioning of Nuclear Facilities in France: Radioactive Waste Management. Proc. NEA/IAEA/EC Workshop on the Regulatory Aspects of Decommissioning (19-21 May 1999, Rome), Rome: ANPA (1999), p.127-133.
- [9] LEBEDEV V.I., et al., Conceptual Aspects of Decommissioning of the First Unit of Leningrad NPP. Ecology and Nuclear Power, 2001, N1, p. 32-37.
- [10] SIMANOVSKIY Yu. M., et al., An Assessment of Impact of Waterside Storage Site for Reactor Compartments of Decommissioned Nuclear Submarines in the North-West Region of Russia on Environment and Population. Proc. Intern. Conf. on Safety of Nuclear Technologies (St. Petersburg, September 2004). SPb: ProAtom, 2004, p. 415-423.
- [11] SHIDLOVSKIY V.V., et al., Concept of Solid Radioactive Waste Management during Decommissioning of IUGR of SChIP. Ibid, p. 475-477.
- [12] THIERFELDT, S., Release of Materials from Regulatory Control. Proc. IAEA Training Course on Decommissioning of NPPs, 23 June-4 July 2003, Karlsruhe (C7-RER-4.025-008).
- [13] DICUS, G., US Regulatory Experience and Perspective in Decommissioning. Proc. NEA/IAEA/EC Workshop on the Regulatory Aspects of Decommissioning (19-21 May, 1999, Rome). Rome: ANPA (1999), p.77-85.
- [14] BEIGUL V.P., et al., Analysis of Comparative Environmental Safety and Necessity of the Final Isolation of Various Types of RW in Russia. Proc. Intern. Conf. on Safety of Nuclear Technologies (September 2004, St. Petersburg). SPb: ProAtom, 2004, p. 92-97.
- [15] RYBALTCHENKO A.I., et al., Deep Disposal of Liquid Radioactive Waste in the Stratum-Collectors: Safety, Results, Prospects. Ibid., p. 394-398.
- [16] BRYZGALOVA R.V., et al., Problem of Radioactive Waste Disposal in Geological Formations of the North-West Region. Proc. Intern. Conf. on Radiation Safety (9-12 November 1999, St. Petersburg). SPb: Restec, 1999, p.7.
- [17] KOMLEV V.N., et al., On the Perspectiveness of Sites in the North-West Part of Murmansk Region for Disposal of Radwaste and Spent Fuel. Ibid., p. 24-25.
- [18] SERGEEV A.S., BOGDANOV R.V., KOMLEV V.N., Estimates of Geological Rock Formations of the North-West Region of Russia as the Environment for Underground Disposal of Radioactive Waste. Ibid., p. 90-91.
- [19] YAKUSHEV M.F., Problem of Radioactive Waste Disposal in the North-West of Russia and Radiological Risk Issues. Ibid., p. 99-100.
- [20] GORBINOV V.A., et al., Vertical Storage Facility for Radioactive Waste and Spent Nuclear Fuel. Proc. Intern. Conf. on Radiation Safety (24-27 September 2002, St. Petersburg). SPb: Restec, 2002, p. 192-195.
- [21] VISHNEVSKY, Y.G., et al., Improvement of the Safety Regulations in the Management of Radioactive Waste Accumulated in the Liquid Radioactive Waste Water Basins of the IA "Mayk" (Ozersk) the Siberian Chemical Plant (Seversk) and the Mining-Chemical Plant (Zheleznogorsk). Proc. Int. Conf. on Safe Decommissioning for Nuclear Activities (14-18 October 2002, Berlin). Vienna: IAEA, 2002, p. 43-53.
- [22] DUBASOV Yu.V., SOKOLOV V.A., Creation of the First Repository of Radioactive Waste in the Cavern Created by Nuclear Explosions. Proc. Intern. Conf. on Radiation Safety (9-12 November 1999, St. Petersburg). SPb: Restec, 1999, p. 15.
- [23] MELENEV V.G., Disposal of Oil Production Equipment Contaminated with Natural Radionuclides. Ibid., p. 82-83.

- [24] CHEPENKO B.A., et al. Radiological Safety in FPC of Russia: Topicality of the Problem and Recent Developments. Proc. Intern. Conf. on D&D and Nuclear and Hazardous Waste Management (13-18 September 1998, Denver). ANS, 1998, Vol.1, p. 354-360.
- [25] POPOVA A.A., Development and Implementation of Radiation Safety Programme in the Osynsky Production Site after Underground Nuclear Explosions. Proc. Intern. Conf. on Radiation Safety (9-12 November 1999, St.Petersburg). SPb: Restec, 1999, p. 27-28.
- [26] GRIGORIEV V.V., et al. Decommissioning of Research Reactor and Its Radiochemical Laboratories. Proc. 3-d Intern. Conf. on Radiation Safety (31.10-04.11.2000, St. Petersburg). SPb: Restec, 2000, p.235.
- [27] TCHUGUNOV, A.S., NECHAEV, A.F., SHIBKOV, S.N., Method for Processing of Liquid Waste Containing Radionuclides, RF Patent N 97117372 of 28.05.98.
- [28] THUGUNOV, A.S., DOILNITSYN, V.A., NECHAEV, A.F., Methodical and Technical Principles of Liquid Radwaste Treatment, Proc. Second Intern. Conf. on Radiation Safety, (St. Petersburg, November 9-12 1999), p. 111.
- [29] TCHUGUNOV, A.S., et al. New Technology for Processing of Concentrates from the Systems of Liquid Radwaste Utilization, Ibid., Suppl. Vol., p.98-104.
- [30] TCHUGUNOV, A.S., NECHAEV A.F., SHIBKOV, S.N., Method of Liquid Waste Purification from the Ions Heavy Metals and Radionuclides. RF Patent N 2127459 of 10.03.1999.
- [31] THUGUNOV A.S. et al. Facility for Electro-Chemical Processing of Liquid Waste. RF Patent N 2004101802 of 27.01.2004.
- [32] SHIBKOV S.N., et al., Pilot Facility for Treatment of LRW by the Electro-Stimulated Destruction at South-Ukrainian NPP. Proc. 5-th Intern. Conf. on Radwaste Management (22-24 November 2005, Moscow). M: VNIIAES, 2005, p. 23-24.

#### **SLOVAKIA**

#### IMPACT OF A1 DECOMMISSIONING WASTE DISPOSAL ON REPOSITORY PERFORMANCE ASSESSMENT METHODOLOGY DEVELOPMENT

## V. Hanušík, E. Hladký

VUJE Trnava, Inc., Okružná 5, 918 64 Trnava

## Abstract

As a part of the Decommissioning project of A-1 NPP, the safety re-assessment study for the Mochovce disposal facility has been finished. The purpose of this study was to update an already existing assessment, in order to demonstrate that an acceptable level of protection of human health and environment can be achieved for the disposal of both operational and decommissioning waste, now and in the future. Estimated inventories and characteristics of potential radioactive waste arising from the decommissioning of A-1 NPP and NPPs with VVER type reactors indicate that are there some significant differences between the decommissioning waste and operational waste, which are likely to have an impact on the safety assessment of the disposal facility. It was decided to follow the ISAM methodology proposed by the IAEA in a safety re-assessment. New FEPs and scenarios were added to the analysis to accommodate the new waste forms in the inventory. This paper summarizes safety assessment aspects of co-disposal of operational and decommissioning waste in Slovakia.

#### 1. SCOPE

As a part of the coordinated research project "Disposal Aspects of Low and Intermediate Level Decommissioning Waste" the research project entitled "Impact of A1 decommissioning waste disposal on repository PA methodology development" was performed by VUJE Trnava under the Research Agreement No. 12185/R0. Specific areas of works relevant to the disposal of A-1 decommissioning waste included:

- --- Establishment of the characteristics and radionuclide inventories of the decommissioning waste from A-1 NPP
- Assessment of the conditioning and packaging options for the various types of wastes
- Safety assessment aspects related to the disposal of the waste in the disposal facility at Mochovce

## 2. KEY FINDINGS

# 2.1. Establishment of the characteristics and radionuclide inventories of the decommissioning waste from A-1 NPP

The nuclear power plant A-1 with a gas cooled  $(CO_2)$  and heavy water moderated reactor (HWGCR) was operated from December 1972 to February 1977. Its average power output was 143 MW(e). There were 148 fuel assemblies and 40 control rods in the core (channel type of reactor). The primary circuit consists of six loops, each with a turbo compressor and a steam generator. The moderator circuit consists of three loops, each with a circulation pump and a cooler.

At the end of February 1977, an accident occurred. As a result of this accident, fuel cladding fracture and fuel uncladding occurred in the upper part of fuel elements over a length of 30

to 100 cm. The primary circuit (coolant) and heavy water circuit were contaminated by fission products and long-life alpha nuclides. Some auxiliary circuits and facilities were also contaminated.

The variety of radioactive wastes increased during the NPP A-1 operation and especially as a consequence of an accident at that plant and subsequent decontamination activities. There are tens of different radioactive waste streams stored inside the NPP A-1 (liquid waste from spent fuel storage, radioactive concentrate, sludges, spent ion exchange resins, radioactive metallic waste, combustible solid waste, etc.).

The main sources of radioactive waste at the NPP A-1 are long-term storage, hot cell, underground storage tanks at water treatment station (complex 41), storage of liquid radioactive waste (complex 44), and primary circuit and heavy water management systems.

## Long-term storage

Long-term storage (LTS) was designed as a pool for cooling and storage of spent fuel during the operation of the NPP A-1. The external cooling medium in the pool was water, while the internal medium inside the carbon steel casings was chrompik (i.e. water solution of  $K_2Cr_2O_7$  or  $K_2CrO_4$  with concentration of 3-5%). Fuel elements immersed in chrompik corroded in their casings.

Having recognized the potential problems with corrosion of spent fuel during long-time storage, the aqueous coolant was replaced by a polyphenyl based organic coolant dowtherm (eutectic mixture of biphenyl and biphenyl-oxide), which was used for the remainder of plant operations. All fuel assemblies removed from the reactor core after accidents were placed in this coolant. Since the corrosion of cladding was negligible in this type of coolant, the resulting activity of dowtherm  $(2 \cdot 10^{12} \text{ Bq})$  was substantially lower than that of chrompik. Corrosion and erosion processes on the surfaces of fuel roads accumulation of radioactive waste at the bottom of all casings with chrompik and in many casings with dowtherm.

Leakages of cooling media from casings resulted in an accumulation of organic sludges at the bottom of the cooling pool. A 15 - 40 cm-thick layer of the sludge covers the bottom of the pool. This sludge is a mixture of organic compounds (mainly dowtherm) and suspension of water and insoluble inorganic compounds. Average <sup>137</sup>Cs volume activity in this sludge is 10<sup>5</sup> Bq.dm<sup>-3</sup> for water phase, 10<sup>6</sup> Bq.dm<sup>-3</sup> for organic phase and 10<sup>9</sup> Bq.dm<sup>-3</sup> for inorganic compounds [1].

# Hot Cell

The hot cell was used during and after the operation of the NPP A-1 for manipulation, cutting and conditioning of damaged fuel elements for safe transport from the plant. Radioactive wastes are mainly in the form of contaminated metals, deposits and sludges. The total volume of waste and sludge is about  $3 - 4 \text{ m}^3$ . The high content of radionuclides in this waste is mainly due to the corrosion and erosion of fuel elements handled in the hot cell.

## **Underground storage tanks (complex 41)**

A complex of six tanks of various volumes (from  $100 \text{ m}^3$  up to  $700 \text{ m}^3$ ) and of various diameters (from 6 m up to 16 m) functioned as storage for mechanically polluted waters from the main building and sedimented waters from laundry. The tanks are not operated anymore. A layer of sludge with a total volume of about  $100 \text{ m}^3$  covers the bottom of tanks nos. 7/1 and

7/2. From a radiochemical point of view these sludges have a high content of alpha-emitting radionuclides  $(1.1 \cdot 10^6 - 7.6 \cdot 10^6 \text{ Bq/kg dry matter of }^{239, 240}\text{Pu})$  and of  $^{137}\text{Cs}$   $(1.3 \cdot 10^9 - 5.8 \cdot 10^9 \text{ Bq/kg dry matter})$  [2].

#### Tanks for storage of liquid radioactive waste (complex 44)

A complex of six cylindrical tanks, each with a volume of 390 m<sup>3</sup>, was designed for storage of liquid radioactive waste from the operation and decommissioning of the NPP A-1. After accidents these tanks were used for treatment of heavy contaminated waters with hexacyanocalium ferrate and copper sulphide. The total volume of sludge at the bottom of tanks nos. 2/1 and 2/2 is about 110 m<sup>3</sup> and the volume of gravel at the bottom is about 100 m<sup>3</sup>. The sludges have a high content of alpha-emitting radionuclides ( $1.3 \cdot 10^6$  Bq/kg dry matter of  $^{239, 240}$ Pu) and of  $^{137}$ Cs ( $3.3 \cdot 10^9$  Bq/kg dry matter) [2]. One tank (N1) with a volume of 35 m<sup>3</sup> was designated to store spent ion exchange resins and sands.

## Concentrates

Concentrates in the NPP A-1 are produced by evaporation of the contaminated water from decontamination, hygienic services, etc. Contaminated water is concentrated in the evaporator at the rate of 1.2 - 1.5 tons of evaporated water per hour. Concentrates are shortly stored in two technological vessels and then solidified. The main compounds are formed by nitrates, oxo/hydroxides of metallic elements, surface-active compounds and by some compounds from spent decontamination solutions. The volume beta/gamma activity of these concentrates is in order of  $10^3 - 10^4$  Bq/l. Salinity ranges between 55 g/l and 65 g/l, pH has value of 8 - 8.5 and yearly production is about 8 m<sup>3</sup> [3].

#### Waste from dismantling of the primary and heavy water circuit

Waste from decontamination and dismantling of primary circuit and heavy water management systems will be typical for the second period of the NPP A-1 decommissioning. The primary circuit consists of reactor, turbo-compressors, steam generators, section valves and pipes of the primary circuit. Primary circuit equipment had been manufactured from low alloy steel (Cr and Ni between 1 and 3 %). The construction material of heavy water circuit and auxiliary systems that were in contact with heavy water is stainless steel. It is estimated that the total gamma contamination of primary circuit is on the order of  $10^{14}$  to  $10^{15}$  Bq; the total alpha contamination on the order of  $10^{11}$  to  $10^{13}$  Bq. The total amount of deposits in the gas circuit is about 14.3 tons [4].

Estimates of amounts and radioactivity in NPP A-1 decommissioning waste are presented in Table 1. The largest masses/volumes of waste are represent by non-processible waste-concrete rubble, contaminated soil, metallic components and compactable waste.

The total radioactivity estimated for NPP A-1 decommissioning waste is approximately 4.0  $10^{15}$  Bq. The main contributors to the inventory include chrompik, sludges, components from dismantling of reactors and sorbents. These wastes comprise a small volume of decommissioning waste. The principal radionuclide is <sup>137</sup>Cs (3.5  $10^{15}$  Bq); other short-lived nuclides include <sup>63</sup>Ni (2.9  $10^{14}$  Bq), <sup>90</sup>Sr (1.2  $10^{14}$  Bq). Of the long-lived radionuclides, <sup>59</sup>Ni (3.4  $10^{12}$  Bq), <sup>239</sup>Pu (2.0  $10^{12}$  Bq), <sup>94</sup>Nb (1.2  $10^{12}$  Bq) and <sup>99</sup>Tc (8.0  $10^{11}$  Bq) are the most important.

# 2.2. Assessment of the conditioning and packaging options for the various types of wastes

The Bohunice Conditioning Centre is the basic facility for final treatment and conditioning of radioactive waste. The following facilities are operated at the Centre:

- Concentration plant for the volume reduction of liquid non-combustible waste,
- Incineration plant for the volume reduction of solid and liquid combustible waste,
- High-pressure compactor for the volume reduction of solid compactable waste, and
- Cementation plant for the conditioning of liquid concentrates and other liquid waste such as ion exchange resins and sludges

Two bituminization plants with a capacity of 120 dm<sup>3</sup>/hour each are operated at Bohunice site. Compounds from the evaporator concentrates from NPPs V-1, V-2 and from the NPP A-1 are fixed in the A-P 80 soft type bitumen and contribute to 40% and 25-30% of the final product respectively. The final product of waste bituminization is placed into 200 dm<sup>3</sup> drums.

The cementation plant is used for macro-encapsulation of the 200 dm<sup>3</sup> steel drums. Drums are filled with waste fixed in a bitumen or cement matrix, solid items of non-compactable and non-combustible waste. Drums and pellets produced by the supercompactor are placed in a standard reinforced fiber concrete container (RFCC). Void space in the container is filled with cement mortar or cemented liquid waste.

New technologies were commissioned in connection with processing of wastes from the NPP A-1 decommissioning:

- Vitrification is used for solidification of radioactive chrompik into a glass matrix of boric-silicate type
- --- Sludges from dowtherm and from the long-term storage pond and tanks are encapsulated in cement or inorganic SIAL matrix directly in the drum
- Pre-treatment of ash by homogenization with paraffin in drums before high pressure compaction

Waste from decontamination and dismantling of the primary circuit and metal waste from demolition of the NPP A-1 reactor will be loaded into a container and grouted with non-active cement mortar. Metal components may be melted, with the resulting slag treated as radioactive waste.

# **3.** SAFETY ASSESSMENT ASPECTS RELATED TO THE DISPOSAL OF THE WASTE IN THE DISPOSAL FACILITY AT MOCHOVCE

A pre-operational safety assessment report was prepared following the request of the Nuclear Regulatory Authority and IAEA WATRP mission by VUJE Trnava during 1998 [4]. This report was based on the US approach to safety documentation for a near surface disposal facility. The choice of scenarios followed the NUREG 1199 standard. Scenarios developed in the guide for trench repositories were adapted to the vault disposal conditions, and were extended to include intrusion scenarios. The main waste stream that was assumed in this safety assessment was low-level operation waste. Only the least active waste of Bohunice A-1 reactor with a total volume 2440 m<sup>3</sup> was addressed in this safety assessment. The main

purpose of the safety assessment was to derive activity limits (total for the whole site, maximum for the individual waste package and maximum averaged for the individual vault).

Approved Limits and Conditions of the Mochovce disposal facility contain a definition of acceptable forms of waste packages. A set of disposal waste packages had been approved by the regulatory authority on the basis of the safety analyses. The following types of waste packages can be disposed in Mochovce repository:

- Containers filled homogeneously by cemented concentrates,
- Containers with drums of bituminized liquid wastes, with void spaces filled by cement mortar,
- Containers with drums of waste metal pieces, with void spaces filled by cement mortar
- Containers with pellets from high pressure compaction of solid non combustible waste, with void spaces filled by cement mortar

These acceptable waste package forms reflect issues concerning suitable conditioning technologies. The waste producer is seeking permission from the Nuclear Regulatory Authority to emplace new types of waste streams arising from NPP A-1 decommissioning works and prepared by new conditioning technologies. According to the view of regulatory authority, all changes in Limits and Conditions must be justified by safety analyses. Recently, a safety re-assessment study for Mochovce facility has been completed [5].

## 3.1. Safety assessment methodology

It was decided to follow the ISAM methodology proposed by the IAEA in a safety reassessment. The key components of the ISAM safety assessment approach consist of:

- the specification of the assessment context (Step 1);
- the description of the disposal system (Step 2);
- the development and justification of scenarios (Step 3);
- the formulation and implementation of models (Step 4); and
- the calculation and derivation of activity limits (Steps 5 and 6).

## Assessment context

The purpose is to update an already existing assessment, to demonstrate that an acceptable level of protection of human health and environment can be achieved for operational and decommissioning waste both now and in the future.

The effective dose to a member of the public as a consequence of an evolution scenario must not exceed 0.1 mSv/year, and as a consequence of the intruder scenario, 1 mSv/year. Time periods for post-institutional control calculations were unconstrained in order to allow for the determination of a peak dose for each scenario assessed.

## System description

The present disposal structures consist of 80 concrete vaults. 90 (3x10x3) standard concrete containers can be inserted into each particular vault, resulting in a total present capacity of the disposal facility being 7200 containers with a total volume of 22,320 m3. The vaults are constructed on a drained gravel layer. Surrounding the vaults and beneath the gravel layer is

an artificial compacted clay layer, which provides an additional barrier against the potential loss of radionuclides The thickness of the clay wall and bottom are 3.5 m and 1 m, respectively. After filling the disposal area, a multi-layered engineered cap with a total thickness of 4 m will be emplaced over the disposal vaults and will include 1.5 m of rolled clay and drainage layer.

#### Waste and waste forms

The total inventory of radioactive wastes (from current and future operations and decommissioning) of all Slovak NPPs disposed of in the Mochovce repository was estimated at the beginning of the safety re-assessment. The inventory consists of two parts-an amount of radioactive wastes represented by a certain number of packages, and an activity represent by 19 radionuclides. A key characteristic of the decommissioning waste, relative to the operational waste, is the predominantly larger inventory of noncompactable metallic constituents. The principal contributors to the total radionuclide inventory include fission products and actinides (<sup>137</sup>Cs, <sup>90</sup>Sr, <sup>99</sup>Tc, <sup>129</sup>I, <sup>241</sup>Am, <sup>239</sup>Pu) and activation products (<sup>63</sup>Ni, <sup>59</sup>Ni, <sup>94</sup>Nb, <sup>14</sup>C) from decommissioning of NPPs.

Assuming co-disposal of decommissioning waste with operational waste significantly increases the final disposal volume. The current volume capacity of the Mochovce repository (7 200 containers) will not be sufficient to accommodate both the operational and decommissioning waste inventory. Disposal of all radioactive waste requires repository enlargement.

## **Development and justification of scenarios**

A systematic, generic list of all possible features, events and processes (FEP) predictable for surface LIL waste disposal was prepared on the basis of the ISAM FEP list. The possible FEPs were screened in a well documented and transparent manner and finally the FEPs, which could significantly influence the performance or safety of the Mochovce repository, were selected.

There are some important differences between decommissioning and operational waste that will likely have an impact on the safety assessment of the Mochovce disposal facility. The FEPs were examined to determine whether the FEPs included in safety assessment (i.e. screened) would account for the decommissioning waste. New FEPs were added to the analysis to accommodate the new waste forms in the inventory:

- Gas generation
- Release mechanisms and radionuclide migration
- Releases by human intrusion

Based on the FEP analysis and regulatory requirements, the following scenarios were developed:

- Evolution scenario
- Combination Well evolution scenario with a well drilled next to the repository
- Inadvertent human intrusion after the institutional control
- Gas release scenario

The evolution scenario includes a succession of processes beginning with a fully operational cover, undergoing the bathtubbing effect until final failure of the barrier containment structures. The human intrusion scenarios included construction scenarios involving excavation of part of the repository. It is assumed that the reinforced concrete containers will ensure safe mechanical protection against possible inadvertent intruders for 500 years after closure.

# **3.2.** Evolution scenario analysis

The evolution scenario analysis consists of three components: near field (source term), far field (groundwater transport) and biosphere. The software utilized for the near field and biosphere is GoldSim. Visual MODFLOW is used as a modelling environment to build and calibrate a two-dimensional model of ground water flow beneath the disposal site to the nearest lake.

## Near field analysis

Some specific near-field modelling issues were considered in the conceptual model. One of the issues is the spatial discretization of the repository. Its key physical components are a clay cover, concrete roof and floor, concrete containers, clay bath and waste forms. Time discretization takes into account the time-dependence of key parameters of these components such as infiltration rate and sorption coefficient due to degradation. In the repository, water flows sequentially through these components. Prior to the failure of the concrete containers, flow occurs through the engineered structures; water will not infiltrate into the waste container. However, once the container starts to fail, water enters the container and contacts the waste form, allowing radionuclides to be released. A distributed failure of different containers was assumed.

The near-field conceptual model assumes five waste forms-cement, bitumen, SIAL, glass and compacted/non-compacted waste. To address contaminant release from these waste forms, the safety assessments used three release processes-diffusion, rinse and dissolution. The rinse release model was used to simulate the release from compacted and uncompacted (metallic pieces) waste. The diffusion release model was used to simulate the release of contaminants from solidified waste (cement, bitumen and SIAL matrix). Vitrified waste is assumed to release contamination into the infiltration water through the dissolution of the glass.

Radionuclide inventories were assumed pseudo non-homogeneously distributed among waste. The waste inventory is representatively distributed among the various waste forms present in the repository. Multiple simulations were performed on each waste form type, and the outputs of the individual simulations were summed as output to the groundwater.

Two gas generation mechanisms were implemented in the safety assessment: microbial degradation of organic components and anoxic corrosion of iron based metals. In this regard, safety assessment calculations were carried out to address the impact of potential gas generation on repository safety including radiological hazard.

## Groundwater transport

The hydrological transport model was built according to previous site investigations and measurements. The model is heterogeneous with respect to hydraulic conductivity. The remaining input parameters to the model were rainfall, effective porosity, drains conductance and longitudinal dispersivity. Most of these parameters were obtained from previous

investigations and were changed slightly in the calibration process particularly because of unsaturated zone influence. The next step was the model calibration, which involved comparison of simulated hydraulic head to measured head data from the year 1995 to 2004.

The hydrological model described above is used to generate a steady-state flow field for the Mochovce site. This flow field is utilized in the transport model MT3DMS to predict movement of radionuclides from their area of contact with the water table through the groundwater to the points of interest.

## Biosphere

Radionuclides released into groundwater can reach surface water bodies. The resulting activity in the water phase causes human exposure due to consumption of drinking water and agricultural products contaminated by irrigation practices. The discharge of groundwater to the biosphere is a long-term process. Therefore, a simple dosimetric model based on the concentration factor methods was used. The dose estimate for a particulate radionuclide was calculated based on the peak ground water concentration and volume of water used.

## **3.3.** Intruder scenario analyses

After the institutional control period, human activities such as the construction of roads and buildings (dwellings) are assumed to take place on the site. In the case of dwellings either simple houses or multi-storey blocks may be built. In the residence scenario it is assumed that after the construction of a building, people live in and around the building, the surrounding of which have been contaminated by the excavation work associated with the construction.

A probabilistic approach was used and uncertainties associated with parameters, processes and events were expressed using probability distributions. The outputs, effective doses, are also probability distributions that represent the uncertainty associated with that estimate. 95th dose percentiles were used for determining concentration limits.

# 4. CONCLUSIONS

From a physical-chemical point of view, waste from the NPP A-1 presents a very heterogeneous composition. The main contributors to the inventory from decommissioning of NPP A-1 include chrompik, sludges, components from dismantling of reactor and sorbents. These wastes comprise a relatively small volume. The principal short-lived radionuclides are <sup>137</sup>Cs, followed by <sup>63</sup>Ni and <sup>90</sup>Sr. The principal long-lived radionuclides are <sup>59</sup>Ni, <sup>239</sup>Pu and <sup>94</sup>Nb.

The safety assessment revision of the Mochovce repository includes disposal of a wide variety of waste types from the operation and decommissioning of all NPPs in Slovakia, and is based on the ISAM approach. The safety analysis demonstrates that the facility satisfies the objectives of protecting human health and environment and meets applicable regulatory requirements. Evolution (groundwater) and intrusion scenarios were assessed.

A standard approach based on estimated total disposed inventory was used for the groundwater analysis. Of the evolution scenarios, the well scenario is critical. The dose to adults reaches a maximum after about 7500 years, but does not exceed 52 percent of the radiological limit. The major contributor to the dose is  $^{129}$ I (97 percent).
Using conservative assumptions, the radiological consequences of prompt transfer of bulk gases containing <sup>14</sup>C has been estimated in terms of the maximum annual dose associated with this radionuclide. The dose,  $1.5 \ 10^{-6} \ \text{Sv/yr}$ , indicates that the potential radiological consequences are not significant. Potential gas production would not have any adverse effects on container or repository performance with respect to pressure build up.

Concentration limits based on intruder scenarios were derived from the dose estimates. The volume of the cover layer of the repository determines the doses from the residence scenario and has influence on the concentration limits for upper layer of the waste packages. Building scenarios limit the concentrations of the radionuclides in central and lower layers of the packages. Accordingly, each radionuclide has a different concentration limit for the upper and middle/bottom layers of the waste packages. The 4000 Bq/g alpha nuclide (<sup>238</sup>Pu, <sup>239</sup>Pu, <sup>241</sup>Am) limit of is more restrictive than the intruder concentration limit calculated in the analysis. In this case, the alpha limit takes precedence.

On the basis of the safety assessment it can be concluded that the Mochovce site has both the volumetric and radiological capacity to accept LIL waste from all NPPs. The current building volumetric capacity of the Mochovce repository (7200 containers) is sufficient only for disposal of operational wastes. Co-disposal of operational and all decommissioning wastes requires a volumetric capacity of about 19 300 containers. Waste from decommissioning of NPP A-1 requires a repository capacity of 5855 containers.

Waste	Amount	Radionuclide activity [Bq]						
	m <sup>3</sup>	<sup>59</sup> Ni	<sup>63</sup> Ni	<sup>90</sup> Sr	<sup>94</sup> Nb	<sup>99</sup> Tc	<sup>137</sup> Cs	<sup>239</sup> Pu
Chrompik	12.8	2.5E+09	3.4E+11	1.0E+14	2.6E+08	7.6E+11	3.0E+15	3.2E+11
Dowtherm	23.5	1.5E+06	2.1E+08	5.7E+10	1.6E+05	4.5E+08	1.8E+12	1.9E+08
Liquid	127.1	7.6E+05	1.0E+08	2.4E+10	8.0E+04	2.3E+08	9.1E+11	3.9E+07
Sludges	301.3	6.4E+10	1.4E+13	3.8E+12	9.0E+11	1.2E+09	2.8E+14	1.3E+11
Sorbents	12.6	1.3E+09	2.8E+11	1.4E+12	1.7E+10	2.2E+09	9.7E+13	3.5E+10
Concentrate	720	1.5E+07	1.6E+10	3.1E+09	1.6E+06	9.9E+08	9.7E+11	1.2E+10
Compactable	5042	1.7E+05	7.8E+08	4.6E+09	8.6E+04	2.2E+09	4.0E+11	4.3E+08
Soil, gravel	1072	6.8E+04	3.2E+08	4.9E+08	7.1E+03	3.1E+06	5.4E+10	3.3E+07
Ash	3	6.1E+03	9.9E+06	1.4E+09	1.1E+06	1.2E+06	3.4E+09	1.9E+06
Oil	40.2	9.6E+02	1.3E+05	4.1E+07	1.0E+02	3.3E+05	1.3E+09	4.2E+06
Abrasions	65	4.1E+04	1.9E+08	3.0E+08	4.3E+03	1.9E+08	3.3E+10	1.8E+08
Air filters	672	6.3E+04	3.0E+08	3.7E+08	6.6E+03	2.3E+08	4.0E+10	2.7E+08
	tons							
Combustible	111	9.1E+04	2.2E+09	6.2E+08	1.5E+05	6.5E+07	4.2E+10	5.6E+06
Concrete,								
soil	8663	1.0E+05	4.8E+08	6.9E+08	1.1E+04	4.7E+08	8.0E+10	9.9E+06
Metal waste	3237	1.1E+09	1.8E+12	1.3E+13	2.0E+11	2.8E+10	3.3E+13	1.2E+12
Reactor								
components	216	3.3E+12	2.7E+14	8.3E+10	1.1E+11	6.6E+08	2.6E+12	1.0E+11
Slag from								
melting	854	1.5E+06	2.5E+09	3.6E+11	2.9E+08	3.0E+08	8.5E+11	5.5E+09
Solidified	713	2.8E+08	3.5E+10	4.5E+12	8.6E+08	1.2E+10	5.8E+13	1.7E+11

Table 1. Estimates of amounts and radioactivity in NPP A-1 decommissioning waste

1 includes solidified chrompik in glass, sludge solidified in cement or silicates (SIAL)

#### REFERENCES

- (1) MAJERSKY, D., et al., Characterization and treatment experience of TRU sludges during decommissioning of NPP A-1 in Slovak Republic, ICEM03, Oxford (2003)
- (2) HOLICKA, Z., et al., Analysis of sludges from tanks complex 41 and 44, Technical report VUJE (2000)
- (3) BREZA, M., Systematically approach to the radioactive waste management in Slovakia, Radioactive Waste Management and Environmental Remediation, ASME (2001)
- (4) Pre-operational Safety Report of the low-level radioactive waste repository in Mochovce, VUJE Trnava (1999)
- (5) HANUSIK, V., et al. Re-assessment of Safety Report of the low-level radioactive waste repository in Mochovce, VUJE Trnava (2005)

#### **SWEDEN**

#### PERFORMANCE ASSESSMENT OF A PLANNED NEAR-SURFACE REPOSITORY FOR SHORT LIVED WASTE FROM DECOMMISSIONING OF SWEDISH NUCLEAR POWER PLANTS

#### J. Carlsson

Swedish Nuclear Fuel and Waste Management Co (SKB)

#### Abstract

The objective of the project is to include short-lived waste from decommissioning of the Swedish Nuclear Power Plants into the existing repository, SFR. Today (2006) the repository is licensed for short-lived low- and intermediate level waste from operation and maintenance of the power plants. The scope of the feasibility study is to evaluate the impact of extending the existing repository to also include decommissioning waste.

The decommissioning waste is similar to the operational waste with a focus on contaminated scrap metal and concrete. Considering the present plans of operating the power plants for 40 years and then following an early dismantling, the extension should be in operation around the year 2020.

The total volume of radioactive short-lived decommissioning waste from the 12 Swedish nuclear power plants was estimated to 150 000 m<sup>3</sup>, mainly low level waste packed in freight containers. Some large components e.g., steam generators and reactor pressure vessels are predicted to be handled without packaging. Some waste from the decommissioning will be regarded as intermediate level. This waste may be disposed of in remaining space in the existing part of the repository, preferably the silo, which has the most "sophisticated" engineered barriers. In order to allow for such a mixture of waste, a new licensing of the extended repository has to be granted.

The existing repository was licensed the year 1988 based on a dose-criteria of < 0.1 mSv/y, to a person in the critical group. Today's licensing of new facilities is based on risk criteria, with risk  $< 10^{-6}$ /y.

The safety analyses of the extension of the repository used the AMBER code. The first step was to replicate the analyses that were performed using other codes for the existing repository. Once this was completed, the model was extended to include the caverns for decommissioning waste. A number of closing options were also analyzed. The result indicates that the safety requirements of the extended repository could be met with simple disposal tunnels (so called BLA tunnels). Introducing chemical and engineered barriers will further reduce the peak release during the first few years after closure.

### 1. THE SWEDISH SYSTEM

Sweden today (year 2006) has 10 commercial nuclear power plants in operation (7 BWRs and 3 PWRs) at three sites along the seacoast, see Figure 1. These nuclear power plants produce almost 50 percent of the electricity used in Sweden. The time schedule and conditions for shut down are still under review. One reactor at the Barsebäck site (BWR) was shut down in 1999 for political reasons. The second reactor was finally shut down the year 2005. The shut down was completed before 30 years of operation. Based on technical (safety) and economic considerations, the operation of a reactor may continue 40 to 60 years or longer. The 12 Swedish commercial NPP's were commissioned in a short time span, from 1972 to 1985. Consequently, operations should be terminated between the year 2012 and 2025, assuming an

operating lifetime of 40 years. Compensating for the loss of electricity production after the nuclear power plants are shut down has not yet been solved.



Fig. 1. Location of Swedish Nuclear Power Plants.

Similar to the waste from operation and maintenance of the power plants, radioactive waste from decommissioning will be treated by the utilities to a form suitable for transport and disposal. Disposal of most of the radioactive waste will be in common facilities owned and operated by SKB, Figure 2. Some very low-level and short-lived waste could be disposed of in local disposal sites. No major decommissioning activities will be performed until disposal facilities exist for the majority of the radioactive waste. Some long-lived waste, mainly reactor internals, could be placed in interim storage some years before disposal. For this type of waste, suitable interim storage facilities have to be available before decommissioning. In addition, the handling of waste packages is easier after some period (years) of decay since waste defined as long-lived also contains significant amount of short-lived radionuclides.



Fig. 2. Swedish waste management system.

The engineering and licensing process for disposal of the radioactive waste from decommissioning requires planning on national level. This planning must be performed in cooperation between the utilities and SKB.

Short-lived decommissioning waste is planned to be disposed of in an extension to the existing repository at Forsmark, the SFR. The repository is situated in crystalline rock at about 50 meters depth, under the bottom of the Baltic Sea. The extension is scheduled for operation from the year 2020, at the earliest. Long-lived waste, mainly material that was close to the reactor core, is first placed in interim storage before being disposed of. This repository is assumed to be similar to SFR but at a greater depth. The final repository for long-lived waste is assumed to be in operation around the year 2045.

### 1.1. Planning for decommissioning

There is no national policy requiring a certain starting point or endpoint for decommissioning. When the dismantling will start is a decision to be made by the owner of the power plant. Only when radiation protection or safety hazards exist may the Swedish authorities require an earlier decommission. Since no major decommissioning project has been performed so far, the conditions to be achieved have been given by the Authorities on a case-by-case basis. It is the responsibility of the owners of the power plants to plan and to execute the decommissioning of their plants. The responsibility to manage its own waste is stipulated in the Act on Nuclear Activities and is included in the license to operate a nuclear facility. Generic decommissioning studies are performed by the Swedish Nuclear Fuel and Waste Management Co, SKB. The company is commonly owned by the Swedish utilities with the task of carrying out the utility's responsibilities for managing waste and estimating costs associated with management of spent fuel, waste and decommissioning of their facilities

Following the final shut down of a reactor and removal of spent fuel, it is assumed that the dismantling will commence, preferably after the process systems have been thoroughly decontaminated. In addition, a period of 5 - 15 years might be required to allow for the decay of short-lived radionuclides. Whether this is required, and how long this period will be, must be decided by the utilities for each nuclear power unit from an ALARA standpoint.

The goal of decommissioning is to be able to use the site for other industrial purposes without radiological restrictions.

# 1.1.1. Repository for short-lived decommissioning waste

Plans currently exist for expansion of the existing SFR facility to enable the disposal of radioactive waste from the decommissioning of the nuclear power plants. Although such proposals were not included within previous safety analyses, some consideration has been given to the potential design and performance of an SFR-3 facility.

# Layout of SFR-3

The planned extension of SFR has a capacity for  $150,000 \text{ m}^3$  of decommissioning waste in new horizontal disposal tunnels. The layout of the tunnel system is given in Figure 3. The horizontal extension of the extended SFR-3 and the fracture zones are given in Figure 4.



Fig. 3. Layout of the extended SFR.



Fig. 4. Extended repository considered in preliminary analyses.

The radionuclide inventory is shown below in Table 1, with that of previous studies for SFR-1 included for comparison. It can be seen that, except for H-3 and Fe-55, the estimated inventory for SFR-3 is below that for SFR-1.

Radionuclide	Half-life	SFR-3	SFR-1, BLA	SFR-1, Total
	(y)	(Bq)	(Bq)	(Bq)
H-3	12.3	2.00E+14	6.6E+08	6.2E+11
C-14	5.73E+03	6.00E+10	3.9E+11	2.6E+13
Cl-36	3.01E+05	2.00E+08	8.2E+07	5.1E+10
Ar-39	268	6.00E+11	-	-
K-40	1.25E+09	6.00E+08	-	-
Ca-41	1.03E+05	2.00E+11	-	-
Fe-55	2.73	2.00E+15	9.5E+10	6.5E+14
Ni-59	7.6E+04	1.00E+12	3.9E+10	2.4E+13
Co-60	5.27	7.00E+14	1.0E+12	1.9E+15
Ni-63	100.1	1.00E+14	6.2E+12	4.0E+15
Sr-90	28.8	3.00E+11	3.6E+11	2.6E+14
Nb-94	2.03E+04	7.00E+08	3.9E+08	2.4E+11
Tc-99	2.11E+05	4.00E+08	4.1E+10	2.6E+13
Ru-106	1.02	1.00E+10	1.5E+05	2.9E+11
I-129	1.57E+07	3.00E+07	2.5E+06	1.5E+09
Cs-134	2.06	1.00E+12	6.3E+09	1.1E+14
Cs-135	2.30E+06	3.00E+07	4.1E+07	2.6E+10
Cs-137	30.1	7.00E+12	3.7E+12	2.7E+15
Pu-238	87.7	2.00E+09	8.5E+09	3.0E+12
Pu-239	2.41E+04	8.00E+08	8.5E+08	3.0E+11
Pu-240	6.56E+03	1.00E+09	1.7E+09	5.9E+11
Pu-241	14.4	6.00E+10	8.8E+10	3.2E+13
Am-241	432	2.00E+09	2.5E+09	6.1E+12
Cm-244	18.1	2.00E+08	3.3E+09	1.2E+12

# Table 1. Inventory in SFR at closure

# 1.1.2. Plugging of tunnel system

The aim of backfilling, plugging and closure of the SFR is to make sure that the safety requirements are fulfilled.

To seal off SFR the following components will be used:

- The storage rooms will be sealed with plugs that provide both mechanic and hydraulic cut off.
- The access tunnels will also be sealed off with plugs that are similar in design to the storage room plugs.
- The storage rooms will be backfilled, providing mechanical support for the rock.
- Some parts of the tunnel system, mainly the unplugged access tunnels, will be backfilled with crushed rock to hinder future intrusion.

The principal design of the access drift plugs and storage room plugs consists of a bentonite section that acts as a hydraulic cut off, and a concrete section that takes the mechanical load and confines the bentonite section. The access tunnels have also been excavated using standard drill and blast technique, requiring the damaged zone to be removed by seam drilling or other suitable technique. The position, width and length of the bentonite section will be determined based on the local hydrogeology and on investigations on the extension of the excavated damaged zone. In this case, the concrete plug can either be keyed into the rock by creating a slot in the same manner as the damaged zone, or the concrete section can be made

longer and convey the load to the rock by friction between the concrete and the rock. This case is illustrated in figure 5.



Fig. 5. Principal design of a plug.

# 1.1.3. Safety analyses

The bedrock surrounding the SFR consists of Leptites and fine-grained gneisses, granites and pegmatites. Within the Forsmark area, the bedrock is fractured in a general block-like pattern. The fractures are dominated by steeply and gently dipped fractures which are orientated in NW and NE directions. The structural geological interpretation has been divided into regional and local scales. The regional scale included only the large regional fractured zones whereas the local scale represents the area immediately surrounding the facility.

### Near-field

Given the limited information available concerning the design and performance of SFR-3, for the purposes of these calculations SFR-3 will be represented using conceptual models from previous analyses configured in AMBER for the BLA.

This approach is considered to be reasonable at this stage for the following reasons

- The calculations are considered to be scoping in nature.
- The wastes in SFR-3 and BLA are both disposed of in containers for which no particular performance as a barrier is required.
- The radionuclide levels in the waste in SFR-3 are most likely to be those of waste disposed of to the BLA.
- --- No detailed tunnel design is currently available for SFR-3 and it is therefore considered that the most simplistic design is suitable for the decommissioning wastes.
- Although SFR-3 is considered in earlier studies, the detailed model results, which are necessary to derive flow fields for the individual tunnels, do not include SFR-3.
- The result of this will be a similar radionuclide transfer rate from the near-field to the BLA.

A schematic figure of the BLA conceptual model used in other calculations is shown in Figure 6. The model used for BLA was divided into waste sections surrounded by a section with only water in each short side (Figure 7). The barriers in BLA are limited and in the model no barriers are taken into account. The radionuclides are initially free to be transported with the water to the surrounding rock.

For comparison some calculations have also been performed assuming the SFR-3 design will be close to that of BMA. In this case the waste is assumed to be conditioned in cement and that the waste packages are disposed of in concrete cells covered by concrete lid and surrounded by crashed rock or sand. Sorption of radionuclides on cement due to the high pH in the near-field is taken into consideration.



Fig. 6. Conceptual model of radionuclide release from the BLA.



Fig. 7. BLA model.

# Geosphere

The geosphere is not considered within these initial scoping calculations, except as a groundwater flow restrictor.

### Biosphere

The following two biosphere models have been considered within these calculations

- Reasonable Development Biosphere (land uplift and change of biosphere from the sea to an agricultural land)
- Today's Biosphere



Fig. 8. A scheme of Reasonable Development Biosphere model.



Fig. 9. A scheme of Today's Biosphere model.

# 2.1 Results

Figure 10 shows a comparison of near-field release for SFR-3 with SFR-1. The release profiles for SFR-3 and the BLA show similar trends due to the similarity of the models they are calculated from. SFR-3 has a larger radionuclide flux than the BLA due to its higher inventory.

Until around 4000 AD, the radionuclide release from SFR-3 is estimated to be larger than any of the facilities within SFR-1. This is due to simple containment being assumed, resulting in large release rates. However, the very long-term release rates from SFR-3 (and the BLA) are estimated to be much smaller than those estimated for the Silo and BMA.



Fig. 10. Comparison of total estimated radionuclide flux for SFR-3 and SFR-1.

Figure 11 shows a comparison of doses to today's biosphere following release from SFR-3 with those arising from SFR-1. As noted previously, SFR-3 and the BLA follow similar trends, with the dose from SFR-3 being slightly higher than that of the BLA. The initial dose from SFR-3 is around 1E-06 Sv/y, higher than the SFR-1 facilities, but falls off rapidly as the radionuclide release rate reduces.



Fig. 11. Comparison of doses to today's biosphere.

Figure 12 shows the near-field fluxes from two alternative SFR-3 'concepts'. The first concept that has been discussed to date is that SFR-3, which is similar in design and performance to the BLA (SFR-3 (BLA)). The alternative considered here is the SFR-3, which is similar in design and performance to the BMA (SFR-3 (BMA)). The near-field fluxes are also compared with those of the SFR-1 facilities in Figure 12.

The BMA concept for SFR-3 results in a much reduced initial release of radionuclides but a higher long-term release rate when compared to the BLA concept.

The effect on releases to the biosphere is shown in Figure 13, which shows a complementary plot to Figure 12 but for release to today's biosphere. It can be seen that the BMA SFR-3 concept results in a much reduced dose when compared to the BLA SFR-3 concept. The dose to today's biosphere from the BMA SFR-3 concept is also lower than any of the facilities in SFR-1.



Fig. 12. Comparison of near-field fluxes for alternative SFR-3 concepts with those from SFR-1.



Fig. 13. Comparison of doses to today's biosphere for alternative SFR-3 concepts with those from SFR-1.

#### Well scenario

To illustrate the influence of a well being introduced in the system, some calculations were made in which both drinking water and water used for irrigation of the land were included. The results are shown in Figure 14. The calculations were made for SFR-1, but as before, the results of BLA are also an illustration of the SFR-3.



Fig. 14. Doses to well biosphere due to releases from SFR-1.

### 3. CONCLUSIONS

Due to the limited information currently available and the scoping nature of these calculations, several assumptions were required to be made at this stage.

- The tunnels within SFR-3 were assumed to be of the same design as the BLA.
- Waste was assumed to be emplaced with minimal containment.
- The radionuclide inventory was assumed to be evenly distributed throughout the tunnels.
- Radionuclide transfer rates were calculated using data derived for the BLA (i.e. volumes, groundwater flow rates, diffusion resistances).
- To illustrate the influence of chemical and engineered barriers, calculations were performed assuming a design of SFR-3 similar to the BMA

The estimated near-field release profile for SFR-3 shows similar trends to the BLA (due to commonalities introduced by the above assumptions). The near-field flux from SFR-3 is larger than that of the BLA due to its higher inventory. Initially H-3, Co-60, Ni-63 and Cs-137 dominate the SFR-3 near-field flux, whereas in the very long-term the flux is dominated by inorganic C-14 and Ni-59.

Until around 4000 AD, the radionuclide release from SFR-3 is estimated to be larger than any of the facilities within SFR-1. This is due to the assumed simple containment that is assumed, which resulted in large release rates. However, the very long-term release rates from SFR-3 (and the BLA) are estimated to be much less than those estimated for the Silo and BMA.

The dose to today's biosphere is initially around 1E-06 Sv/y and is dominated by contributions from Co-60, Ni-63 and Cs-137, and in the very long-term falls to below 1E-10 Sv/y (due to releases of inorganic C-14). As expected, SFR-3 and the BLA follow similar trends and the dose from SFR-3 is slightly higher than that of the BLA. The initial dose from SFR-3 is around 1E-06 Sv/y and is higher than the SFR-1 facilities, but falls off rapidly as the radionuclide release rate reduces. At around 2500 AD, the maximum dose is attributable to the BMA, which itself is surpassed by that from the Silo at around 3300 AD.

An alternative SFR-3 concept is also explored in which the design and performance is more similar to the BMA than BLA. In this case the near-field flux in the short-term is greatly reduced when compared with the BLA SFR-3 concept (although the long-term release rate is higher). The corresponding release rates to today's biosphere are also much reduced compared to the BLA SFR-3 concept. These additional calculations therefore show the effect of the inclusion of additional barriers on radionuclide release.

#### UNITED KINGDOM

#### ASPECTS OF PACKAGING METALLIC RADIOACTIVE WASTE FOR DISPOSAL

#### I.H. Godfrey

Nexia Solutions

#### Abstract

Within the nuclear industry in the UK, and world-wide, there is a large amount of metallic material present in decommissioning wastes which will have to be treated for disposal. In particular, the UK needs to treat steels, aluminium, Magnox (a magnesium aluminium alloy) and uranium metals. In the UK the preferred process for the treatment of these wastes is to encapsulate them in a matrix based on ordinary Portland cement, typically blended with blast furnace slag or pulverized fuel ash. As water is present in the cement matrix, even after hydration has occurred, corrosion reactions can take place. This has several significant consequences, which include:

- Possible generation of hydrogen gas from the corrosion reaction.
- Possible generation of expansive corrosion products which may eventually cause degradation of the encapsulation matrix.
- Possible generation of methane and other hydrocarbons formed from the reaction between carbides present in the metallic wastes and water present in the cement matrix.

In the UK the corrosion reactions of metals encapsulated in a cement matrix have been studied for over 20 years; as they are important for the successful encapsulation of reprocessing wastes, such as Magnox fuel cladding, as well as the treatment of the decommissioning and legacy wastes.

This report covers the UK's contribution to the CRP and it aims to review this experience on corrosion of metals in a cement matrix, in particular covering the following areas:

- A general outline of the corrosion behaviour of the different metals.
- A discussion of general corrosion measurement techniques.
- Issues with long term extrapolation of behaviour and accelerated testing.
- Modelling of corrosion reactions and wasteform evolution.

#### 1. BACKGROUND

The nuclear industry in the UK has generated a wide range of wastes at a number of sites over the past 40 years. Wastes have been generated from a range of activities across the nuclear fuel cycle including:

- Fuel manufacture
- Enrichment
- Reactor operations
- Reprocessing
- Decommissioning

The UK has selected ordinary Portland cement (OPC) blended with blast furnace slag (BFS) or pulverized fuel ash (PFA) as the preferred matrix for LLW or ILW waste encapsulation. The benefits of using an OPC cement based matrix include:

- The encapsulation process is a simple, versatile, safe, low temperature process, which uses proven technology and is able to deal with a range of wastes. The process produces minimum secondary waste and has a low operating and capital cost.
- From a waste properties and disposal point of view, cement makes a product which is generally stable to chemical, radiation, thermal, and mechanical effects. Short-lived radionuclides are contained by the matrix whilst the high pH, absorption properties, and low permeability of the matrix minimises the solubility and transport of long lived radionuclides. A large proportion of the UK wastes also contain a significant amount of water which is compatible with a cement based system. The high pH will also minimise internal corrosion of the stainless steel container. In addition, the cement matrix is chemically compatible with the currently proposed UK disposal environment.

Decommissioning wastes, which will have to be treated in the UK and other countries [1] are likely to include significant volumes of various metals, for example in the UK:

- Various types of steel from construction, reactor components, flux flattening bars, control rods, shut off rods, heat exchangers, etc.
- Aluminium from fuel cladding and fittings such as braces, splitters etc.
- Magnox from fuel cladding and fittings such as braces, splitters etc.
- Uranium from metallic fuel, usually associated with the fuel cladding.

One of the issues associated with using a cement matrix for the encapsulation of metallic ILW and LLW waste is that as the matrix is water based, corrosion reactions can take place between this water and any metals present. As this water is derived from the cement pore solution it is of a high pH, typically about 13. This has the advantage of passivating or reducing the corrosion rate of some metals e.g. steels and Magnox. However, for other metals such as aluminium a higher pH will increase the corrosion rate. Despite the passivation of metals, such as Magnox, there is sufficient water present in the pore structure of the set cement matrix to generate some corrosion reactions, albeit often at extremely slow rates, even in a mature matrix. It is important to understand these corrosion reactions and mechanisms for metals encapsulated in cement for several reasons.

Firstly many of the corrosion reactions will produce hydrogen and depending on the metal this can be at significant volumes during the cement curing stage. In addition, there will be a slower long-term generation of hydrogen produced from the metal in the encapsulated set product. This hydrogen evolution is important in two respects:

- --- The safety aspects of the generation of a potentially explosive gas during processing, storage and disposal and the requirement to ensure that the hydrogen is safely dispersed.
- If the rate of hydrogen generation is greater than the rate at which it can permeate through the cement matrix, then internal pressure may build up which eventually overcomes the strength of the matrix causing fracturing. Fracturing is undesirable as it may ultimately increase the rate at which nuclides can be leached from the matrix after disposal as well as reducing the overall integrity of the wasteform.

Secondly, the corrosion of the metal usually produces the metal oxide or hydroxide which is usually a less dense material than the parent metal. Therefore, the corrosion product takes up a larger volume than that occupied by the parent material and hence when corrosion occurs an expansion will take place within the matrix. If this expansion is too great, then eventually it will overcome the ability of the matrix to accommodate the generated strain through either its own strength or through creep, and the matrix may start to fracture. Hence, it is important to understand the long-term rate of corrosion so that it can be demonstrated that the encapsulated waste will have adequate performance throughout the required storage and disposal period. A final issue is that corrosion will generate a source term for the release of radionuclides from the waste.

A large proportion of the UK's reprocessing and decommissioning wastes consists of metallic wastes. Therefore, in order to satisfy the requirements of the UK regulators (NII<sup>\*</sup> and EA<sup>\*\*</sup>) and disposal agency (Nirex) as well as allowing the operation of the existing waste encapsulation plants, waste producers have had to demonstrate a thorough understanding of the reactions, rates and consequences of the corrosion of the metals in their wastes. This is also an ongoing requirement for the development work supporting the treatment, storage and disposal of the UK's decommissioning and legacy wastes though the NDA<sup>\*\*\*</sup>.

The emphasis of this contribution by Nexia Solutions (part of the BNFL group) to the Coordinated Research Project is to review the corrosion behaviour of steels, aluminium, uranium and Magnox metal wastes encapsulated in cements and to identify and discuss the issues associated with producing a satisfactory wasteform.

#### 2. OBJECTIVE OF UK CONTRIBUTION TO THE CRP

The overall objective of the UK contribution to the CRP is to review the corrosion behaviour of steels, aluminium, uranium and Magnox metal wastes encapsulated in OPC based cements and to identify and discuss the issues associated with producing a satisfactory wasteform. This work will contribute to the following specific objectives as stated in the IAEA information sheet for the CRP:

- 'Expand the database on conditioning technologies for the various types of waste'
- 'Assess corrosion-induced degradation of metallic components'
- 'Use appropriate models to predict waste form/package performance over time'
- 'Evaluate the potential for gas generation from metal corrosion and its impact on repository performance'

Whilst the UK contribution will concentrate on wastes encapsulated in cement matrices the principles discussed are likely to be able to be employed for the assessment of corrosion in other matrices.

#### 3. **SCOPE**

The corrosion of Magnox, uranium, steel and aluminium metals in an OPC based cement matrix were selected for discussion as they are the predominant metals of interest in UK

<sup>\*</sup> Nuclear Installations Inspectorate \*\* Environment Agency

<sup>\*\*\*</sup> Nuclear Decommissioning Authority

wastes. It is not intended to discuss the corrosion of the materials of construction of the product containers. It is known from UK studies that the corrosion properties of metals in cement based matrices are very dependent on such parameters as:

- Storage temperature.
- Chemical and physical properties of the encapsulation matrix.
- The exact chemical composition of the waste e.g. different alloys and purity of the metal.
- Shape / surface area of the waste encapsulated.
- Surface condition of the waste clean surfaces or presence of protective layers.
- Galvanic coupling.

Therefore only a general overview of the corrosion properties of the metals will be possible, as data such as corrosion rates and gas generation rates will depend to a large extent on the actual conditions and environment the encapsulated waste will experience. However, broad advice is provided as to how corrosion data might be generated.

4. REVIEW OF EXISTING UK EXPERIENCE

Work on the disposal of ILW started in the UK in the early 1980's [2]. The first material studied which was susceptible to corrosion was Magnox swarf. The corrosion behaviour of Magnox was studied for most of the 1980s, with corrosion data generated, for some samples, for a large proportion of this period. Since the operation of the Magnox encapsulation plant in 1990, further work has been carried out to understand the effects of modifications to the waste feed, such as treatment of stored Magnox, on the quality of the product and also to support the disposal of decommissioning / historic wastes. Aluminium and uranium corrosion in cement has been studied to support the disposal of decommissioning / historic wastes. Less detailed work has been carried out for steel, as it is much less reactive in cement than the other metals assessed and hence is of much lower concern. Typical encapsulated metallic wastes are shown in Figures 1 and 2.



Fig. 1 Encapsulated steel waste.



Fig. 2 Encapsulated Magnox metal waste.

#### 4.1. A general outline of the corrosion behaviour of the different metals.

For magnesium, in the alkaline cement environment, the reaction is simply:

$$Mg + 2H_2O \rightarrow Mg(OH)_2 + H_2$$

For aluminium, initially the high pH solution breaks down the protective alumina layer:

$$Al_2O_3 + 2OH^- + 7H_2O \rightarrow 2[Al(OH)_4.2H_2O]^-$$

The aluminium metal then reacts to form a soluble hydroxyl aluminate species and hydrogen:

$$2AI + 2OH^{-} + 10 H_2O \rightarrow 2[AI(OH)_4. 2H_2O]^{-} + 3H_2$$

The hydroxyl aluminate then reacts further with the calcium hydroxide and silicates present in the cement matrix to form strätlingite. An aluminium hydroxide (bayerite) product is also formed as a precipitate from the solution [3].

With uranium and steel the reactions become more complex, with different reactions taking place depending on the environmental conditions, for example whether or not any oxygen is present in the system. For steel in an oxygen environment the simplified reactions can be:

$$4Fe + 6H_2O + 3O_2 \rightarrow 4Fe(OH)_3$$
$$2Fe + 2H_2O + O_2 \rightarrow 2Fe(OH)_2$$

In an oxygen free environment, corrosion can proceed anaerobically by the following reactions:

$$Fe + 2H_2O \rightarrow Fe(OH)_2 + H_2$$
$$3Fe + 4H_2O \rightarrow Fe_3O_4 + 4H_2$$

In the case of steel, corrosion will take place much quicker if oxygen is present.

Like steel, uranium will follow different corrosion reactions depending on whether or not oxygen is present. In the presence of oxygen the nominal overall reaction is:

 $U + O_2 \rightarrow UO_2$ 

In the absence of oxygen, or once all the oxygen has been consumed, the reaction becomes:

$$U + 2H_2O \rightarrow UO_2 + 2H_2$$

With uranium corrosion the reactions shown are greatly simplified as the reaction is non stoichiometric and  $UO_{(2+x)}$  is formed. Also, depending on conditions, uranium hydride can be formed in addition to the oxide and also other oxides can be formed. If hydride is formed further reaction will occur to produce hydrogen and the oxide leaving a few percent of hydride in the reaction product. Unlike steel, the corrosion reactions are much faster for uranium if oxygen is not present.

For all the reactions there is the possibility of further reaction between the metal oxide or hydroxide formed and the cement matrix to form more complex silicate minerals.

In general, the relative reaction rates of the different metals in cement follows the following trends. Initially, aluminium is more reactive than Magnox, which is much more reactive than steels. As the grout hydrates and sets, the reactivity of aluminium rapidly decreases. Magnox however, will continue to react at a slow rate. The reactivity of uranium is more complex as it usually exhibits an incubation period, where no corrosion or very low corrosion rates are observed. This period can persist for several weeks depending on the history of the metal and ambient conditions. In general, once through this incubation period uranium will react faster than the other metals. However, the reaction rate of uranium will still be lower than the initial high rate of aluminium. The corrosion reduces through continued cement hydration, build up of protective layers and reduced rates of water transfer through the matrix. However, for Uranium this decrease can be small as uranium is extremely efficient at extracting water from the cement matrix and, due to its high atomic mass, large weights of metal can react without significantly affecting the water content of the system.

#### 4.2. Factors which can affect the corrosion rate of metallic waste

There are many factors which can affect the corrosion rate of encapsulated metallic wastes. These factors include the water content, water transport, temperature, metal surface condition (pre-corrosion), available surface area and waste geometry, encapsulation matrix, chemical environment pH, voidage, oxygen availability and galvanic coupling.

The total water content of the wasteform has a major effect on the corrosion rate of the metallic waste with higher water contents increasing the corrosion rate. This water can be that associated with the waste as well as the encapsulant. Hence, it can be beneficial in terms of reducing corrosion rates to minimise the total amount of water present. A similar effect is the rate of transfer of water through the encapsulant, where a lower permeability to water could reduce the corrosion rate by preventing sufficient water reaching the metal. An additional factor which can affect the water availability is how well the waste is infilled. Poor infilling can leave voids which provide a reservoir of water in which the metal could be corroding, which will result in the waste corroding at a faster rate compared to that which has been well encapsulated. With time, the water content of the wasteform will reduce as water is consumed by the corrosion reaction and further hydration of the matrix or lost through other mechanisms. As well as these factors the build up of corrosion product layers can reduce the rate at which water can reach the metal surface. All these factors can cause the observed corrosion rate to reduce with time.

Another major factor is the temperature of the wasteform, with higher temperatures having the potential to significantly increase the corrosion rate. It is therefore important when carrying out corrosion studies to store the test specimens at temperatures which realistically represent those the wasteform is predicted to experience. Being too pessimistic and using too high a temperature can result in major overestimates of the rate of corrosion, which generates significantly lower predicted lifetimes. Another important factor to consider, particularly if the early corrosion rates are of interest, is the effect of the curing exotherm on the reaction mechanism. This can have two impacts on the corrosion rate measured, firstly the corrosion rate can be increased due to the higher temperatures but secondly the corrosion rate can also be reduced due to the accelerated curing of the wasteform and the earlier reduction in water content.

The next set of factors to consider all relate to the characteristics of the waste itself. Important characteristics which can affect the corrosion rate are the composition of the metal itself i.e.

purity and alloying, condition of the metal surface and geometry of the waste. In trials it is important to use the correct alloy or metal of the correct purity as these factors will affect the corrosion rate observed. For example, more corrosion products were found to be formed at the grout metal interface for lower purity aluminium compared to high purity aluminium [4]. The condition of the metal surface is of particular importance during the early stages of the corrosion reactions. For example, some metals such as Magnox will corrode much faster if the surface of the waste is clean compared to the surface being examined in a corrosion experiment. Conversely, uranium can show an incubation period before starting to corrode. The length of this incubation period can be weeks long depending on the condition of the metal. Therefore, it is important to have characterized the waste well enough to be able to use the correct surface condition for corrosion trials and to generate realistic results. The geometry of the waste is another important consideration. As the corrosion reaction depends on water being available at the metal surface, then if the packing, or form of the waste prevents, or restricts, the access of water the corrosion rate will be decreased. This effect could be important for well-packed wasteforms.

In addition, whilst not affecting the 'true' corrosion rate, the effective surface area of the waste in a package will determine the total corrosion observed. It is therefore important to understand the total surface area available for corrosion within the encapsulated wasteform so that a reasonably realistic total amount of corrosion can be generated.

Another major set of factors to consider relates to the chemical environment in which the corrosion is taking place. Examples of this are the pH, oxygen availability, and the composition of the matrix. The pH is very important in controlling the reaction rate for metallic wastes such as aluminium, where the corrosion rate will significantly increase at high and low pH. In a 'normal' OPC cement system, aluminium will initially corrode very quickly due to the high pH, however work has shown that encapsulating the aluminium in a calcium-sulpho-aluminate (CSA) cement, which has a significantly lower pH, will greatly reduce the corrosion rate [5, 6]. This CSA cement will also chemically bind up to five times as much water as OPC systems, which could also help reduce corrosion rates. For other metals such as steel or Magnox, a high pH can be beneficial in reducing the corrosion rate. The presence of oxygen can also affect the corrosion rate of metals in different ways, for example steel will corrode faster in the presence of oxygen. The composition of the matrix can affect the corrosion rate through the pH, and oxygen availability as well as water content and water transfer properties.

A final factor which can affect corrosion rates is galvanic coupling where the reactivity of one metal is modified by being in electrical contact with another. It is important to take galvanic coupling into account when designing experiments to understand the corrosion behaviour of mixed metallic wastes.

### 4.3. A discussion of general corrosion measurement techniques.

In predicting the effects of metallic corrosion on the performance of an encapsulated wasteform there are several areas which will need to be understood:

— The corrosion reactions and mechanisms which can occur and how they may vary with the conditions experienced by the wasteform.

- How the products of the corrosion reactions interact with the encapsulation matrix and container and what effect this has on the performance or characteristics of the wasteform.
- The rate of corrosion and hence the rate at which this performance will change or the characteristic will occur.
- What the consequences of this characteristic or change in performance are. For example does it actually make any difference, or where does it become unacceptable?

Corrosion can be measured in different experimental conditions depending on the objective of the investigation. For example, corrosion studies in water or, preferably, simulated pore solution can be used to generate fundamental information on the mechanisms and corrosion products produced. However, to generate corrosion rates in the wasteform an encapsulated simulant will have to be used. There are many techniques that can be used to understand the corrosion mechanisms and how the corrosion products interact with the wasteform, e.g. SEM, XRD. This paper will concentrate on the techniques available for measuring the rate of corrosion.

### 4.4. Techniques for measuring corrosion rate

There are many techniques which can be used for measuring the corrosion rate of encapsulated materials. A useful guide to corrosion including testing techniques, theory and interpretation can be found in the ASTM 'Corrosion Tests and Standards' manual [7] and ASM 'Corrosion: Fundamentals, Testing and Protection' handbook [8]. It is important when selecting which technique to use to ensure that the technique chosen will be measuring a relevant parameter. For example, under certain conditions uranium may corrode with the evolution of hydrogen gas whereas in other conditions little hydrogen is formed. If the experiment is being carried out to measure hydrogen gas generation, for example in use in a safety case, then it is important that the correct conditions are used. It is also preferable, in such a case, that the rate of gas generation is measured rather than measuring the rate of metal loss, which could predict a much higher hydrogen generation rate than would be observed from the real wasteform.

#### 4.5. Weight changes, and direct measurements of corrosion products.

As encapsulated metallic waste corrodes it will typically react to form an oxide or hydroxide corrosion product. If this remains adhered to the metal, then the specimen will increase in weight in proportion to amount of corrosion that has taken place. Knowing the chemical reactions which have taken place and measuring the weight change will allow the corrosion rate to be determined. Weight change measurement will often be of limited use in the study of encapsulated metallic wastes as these tend to be in closed systems. Hence, the only weight changes observed are likely to be from the evolution of gases which will produce a change too small to be realistically measured. Direct measurement of corrosion products, for example the measurement of corrosion product layer thicknesses in a sliced sample, can give some useful information. However, for encapsulated metallic wastes a large number of samples would be required to measure how the corrosion rate changes with time and so the technique is probably best suited to understanding the corrosion products and their interaction with the encapsulating matrix.

#### 4.6. Gas evolution - pressure, mass spec, gas chromatography

A more useful approach is to measure the rate of hydrogen generation from the corrosion reaction. As discussed above it is important to ensure when using this technique, particularly with metals like uranium, that hydrogen will be generated under the conditions of interest. In this technique the hydrogen gas is usually collected in a vessel and the change in concentration measured with time. Techniques which can be used include:

- Pressure increase
- Volumetric gas collection
- Mass spectroscopy
- Gas chromatography
- Hydrogen concentration measurement by a specific sensor

If experimental conditions are well controlled and a large number of measurements are carried out, preferably in real time, these techniques can prove to be sensitive to very low corrosion rates. However, it is very important, particularly if the pressure rise is being measured, to ensure that the samples being studied are in containers which have very low leak rates, as even the smallest leak path can prevent a meaningful measurement of corrosion rate. Another important consideration with this type of measurement, particularly during the early stages, is that some encapsulants and corrosion reactions can absorb oxygen from the system. If significant, this can complicate early corrosion rate measurements. A further complication to be accounted for in analysis is the variability of the water vapour pressure with temperature; this is particularly important if trials are carried out at elevated temperatures.

These techniques have the benefit of allowing the corrosion to be studied from real items of waste as they would be configured in the encapsulation matrix. However, drawbacks are that the technique cannot measure corrosion if the reaction does not generate gas, or unrepresentative rates may be observed if the conditions are set to ensure that hydrogen is generated. This is particularly important if steels or uranium are being studied. Additional issues are that the corrosion may be taking place in an unrealistic atmosphere caused by the vessel being sealed. For example, the oxygen content could become depleted compared to the real waste product or the water vapour is unable to transfer into or out of the system. The consequences of these on the rates measured and their interpretation should be understood when deciding how a corrosion test should be conducted. This issue will apply to some extent to all the measurement techniques discussed, but is particularly applicable to sealed systems.

#### 4.7. Electrical properties

There are a whole series of techniques which involve measuring the change in electrical properties of the metal as it corrodes. These techniques can be useful as they provide direct measurement of the rate of conversion of the metal to the corrosion product in the encapsulation matrix. The various techniques are discussed in detail in references 6 and 7. However, they have a disadvantage in that they require regular shaped specimens to be able to interpret the results. The real waste may have an irregular shape and packing, which could, for example, reduce the water availability and hence possibly reduce the corrosion rate below that observed in the regular shaped specimen.

### 4.8. Acoustic emission

Acoustic Emission (AE) refers to the generation of transient elastic waves within a material as it corrodes. These elastic waves are detected and converted to electrical signals using piezoelectric transducers on the surface of the structure. If a set of piezoelectric transducers are used then the AE source can be located and, thus, the corrosion location area in a bulk can be identified. Hence, AE inspection is a powerful aid to the study of corrosion and associated deformation and fracture and the technique has the potential to monitor the corrosion of encapsulated wastes and generate information about where the corrosion is taking place. The technique has the benefit of being able to measure the corrosion of metals in a 'real' wasteform.

With all the measurement techniques discussed there are several other important considerations. These are:

- Data logging use of a technique which will allow automated data logging to be carried out will generate significantly better corrosion data than a simple manual recording of data. However, given the long time periods over which corrosion studies are performed significant quantities of data can be generated and if sample periods are too short then data files can quickly become unwieldy. Sample periods of less than an hour are not recommended for long term studies.
- Control of environment the corrosion reactions are usually highly temperature sensitive. It is important to ensure that any trials are stored under well controlled and realistic temperature conditions for the waste form under consideration. It is possible to control samples to within  $\pm$  1°C for lengths of time long enough to generate reliable data. Another important consideration is the transfer of water to and from the reaction environment. As the water availability is an important component of the corrosion of metallic wastes it is important that the experimental systems are created with the correct amount of water present and maintain this water content, i.e. do not dry out or absorb water.
- Sensitivity if the corrosion rate changes with time the measurement technique may need to be changed, or modified, to provide the right sensitivity. For example to be able to cope with a high initial rate and a low long term rate.
- Moving samples when samples have been moved a period of increased corrosion has been observed, after which the rate reverts back to 'normal'. Because of this it is important that the movement of samples during experiments is minimised.

### 4.9. Issues with long term extrapolation of behaviour and accelerated testing.

All corrosion measurements carried out under realistic conditions can only be measured for a short period of time relative to the required lifetime of the wasteform. This raises the question of how to extrapolate the data to predict the performance of the real wasteform. It is possible to use samples with accelerated corrosion rates to generate a description of how the performance of the waste form will evolve with time. By monitoring the corrosion of the accelerated samples it is possible to generate a cumulative corrosion at which these performance changes are observed. For example it may take 'x' % corrosion of a waste before the matrix starts to crack. The time it will take a wasteform to reach this degree of corrosion can then be predicted by using the 'normal' corrosion rate measured and extrapolating this rate into the future.

The simplest methods of accelerating the corrosion of a wasteform are by increasing its temperature, increasing the water availability, and including a chemical accelerant such as sodium chloride. Using these techniques it is possible to accelerate corrosion rates so that for some metals a hundred years worth of corrosion can be observed in a few months.

There are several issues with this extrapolation approach which need to be understood to ensure that the results of any extrapolation are correctly interpreted:

- The corrosion rate observed during the experiment is likely to continue to decrease with time as the water is slowly used up. In a wasteform where this water is not replaced then the corrosion rate decades in the future is likely to be greatly overestimated. Taken to an extreme, care must be taken to ensure that when extrapolating for long periods the corrosion rate assumed doesn't use up all the water within the time period of concern; if there is no water present then the waste can't corrode further.
- Accelerating by addition of extra water may hide effects caused by removal of water from the matrix, whereas the addition of accelerating chemicals or increasing the temperature may themselves modify the performance of the wasteform. For example chlorides also accelerate cement hydration thus removing water from the system sooner. Increased temperature can also accelerate cement hydration but can also cause decomposition of phases releasing water which can also complicate the issue. The addition of extra water at the mixing stage will also significantly change the properties of the encapsulating grout, for example the strength will be reduced.
- Accelerating the corrosion rate will mean that certain slow reactions cannot occur within the matrix which could otherwise accommodate this corrosion at a slower rate. Examples of this are creep of the matrix relieving stresses generated by expansive corrosion products or healing of cracks.
- Acceleration of corrosion by increasing the temperature can change the reaction mechanisms which are occurring; an example is for Magnox where the corrosion mechanism changes at about 60°C. This could change the method by which the wasteform degrades. Also, increasing the temperate too high can change the form or nature of the minerals present in the cement matrix.

It is also important to consider that changes in wasteform performance are not likely to be sudden changes from 'x' to 'y' but rather a gradual change from something looking like 'x' to something looking like 'y'. The individual packages within a group will also have a range of properties at a given time as the contents will be corroding at different rates due to slightly differing starting conditions, compositions, and storage conditions. Therefore, at a given time the packages will have a distribution of properties and hence when quoting a 'lifetime' for a package care must be taken to ensure that it is not construed as meaning that all the packages have suddenly changed from being like 'x' to like 'y'.

When assessing the effects of the long term corrosion rate on the wasteform it is important to remember that the slow long term corrosion rate occurring over many years is much more important then the initial very fast rate which only occurs for a short period of time. The effect of the short-term rate is often to only change the long-term predictions by a few years in 100 years, which is insignificant when compared to other variability associated with the prediction.

### 4.10. Modelling of corrosion reactions and wasteform evolution

There are many areas in which modelling can be off benefit to understanding the consequences of corrosion occurring in the matrix. Areas in which modelling can be employed include: crack propagation, effects of creep, effects of expansion on the package, water transfer, effects of curing exotherm, effects of variability in the waste and extrapolation to understand long term performance.

Crack propagation modelling and modelling the effects of creep has been carried out to understand how cracks will form and propagate as corrosion proceeds. This modelling can be used to better understand the pessimisms introduced by accelerated testing to failure. This understanding can allow a correction factor to be introduced which will generate a more realistic prediction of long-term wasteform performance.

Waste packages are usually large items such that it is not practical or desirable to carry out large numbers of full scale corrosion trials. Therefore, much corrosion work is carried out at a smaller scale where the conditions can be better controlled to give better data and the lower costs per trial allow more variables to be studied. Modelling studies can then be carried out on full-scale packages to understand how the ongoing corrosion will affect the waste package. Examples include the extrapolation of the short-term small scale data to the longer term, modelling the effects of expansion of the waste on the structure of the wasteform, and understanding how the container itself will fail. For complex wasteforms containing many different components modelling can be beneficial in allowing the effects of the different combinations of waste compositions to be understood. Gaining this type of understanding purely from experimental studies is often not practical due to the number of trials required. Modelling can enable an understanding to be gained of the interactions of the wastes which then allows specific targeted trials to be carried out to confirm the understanding.

Another factor which is important for the short-term corrosion rate is the curing exotherm; as discussed earlier this can increase or reduce the corrosion rate. There is also a feedback mechanism whereby the heat generated from the corrosion reaction can increase the curing exotherm, which will in turn affect the corrosion rate, etc. Modelling can be of benefit in understanding these effects particularly when combined with modelling of water transfer through the hydrating matrix.

The extrapolation of product performance is also amenable to modelling. A simple approach has been discussed earlier where the lifetime of a wasteform is predicated by extrapolating corrosion rates. Another model which has been developed in the UK is MAGGAS which allows the gas generation from a wasteform to be predicted under various storage and disposal scenarios, many centuries in to the future.

### 4.11. Alternative 'cement type' systems for modifying corrosion rates

As discussed earlier, the corrosion rate will be dependent on the chemical environment of the encapsulating material. This gives the opportunity to tailor and optimize the encapsulant to minimize corrosion. This is particularly useful during the first few days after manufacture of the waste package, where corrosion rates can be significantly higher than in the long term and the hydrogen gas generation can have direct safety implications. For example, a typical BFS/OPC cement system used in the UK will have a pH of 12.5-13 [9] whilst a PFA/OPC cement system will have a slightly lower pH of about 12-13. This can be taken further by using a calcium- sulpho-aluminate cement (CSA) system as is used extensively for

construction in China and if appropriately formulated has a pH of 10-11. The rate of aluminium corrosion increases rapidly at high pH, hence slightly reducing the pH from that of an OPC based system to a CSA based system can dramatically reduce the corrosion rates.

For uranium metal the corrosion rate is dependent on the presence of oxygen; if oxygen is present then the corrosion rate is reduced and hydrogen is not produced. It has been found that by adding barium peroxide or another source of oxygen [10] the uranium incubation period can be extended and hence the early uranium corrosion / hydrogen evolution rate can be reduced.

Reducing the water content of the grout mix can also reduce corrosion rates. This can be achieved by either using superplasticisers or by modifying the particle size distribution [11] of the cement powders if the use of superplasticisers is not allowed. Alternatively, as discussed earlier, using CSA cement could potentially bind five times as much water as a normal OPC based cement system.

# 4.12. Fundamental underpinning science projects being carried out at UK university research alliances

In the UK fundamental research into the waste immobilization is being carried out by the 'Immobilization Science Laboratory' at Sheffield University [12] in conjunction with Nexia Solutions (part of the BNFL group). The first project of relevance is titled 'Corrosion of Metals in Composite Cements'[13] and is investigating the fundamentals of Magnox and aluminium corrosion. Details of this work have been referenced as appropriate in this paper. Work is also being carried out to investigate the performance of and further develop alternative cement matrices such as the CSA discussed earlier.

Another project is investigating the availability of water as a cement matrix cures [14]. This is using nuclear magnetic resonance techniques to measure the proportion of water held up in the matrix as chemically bound C-S-H water, gel water and capillary water. By understanding the fate of the water as the cement hydrates, the water availability in the long term, and hence long term corrosion behaviour may be better understood. Initial studies have monitored the change in pore size, the amount of water trapped in closed capillary pores, and in small C-S-H gel interlayer spaces in cement samples cured for up to a year. These have been compared with measurements made by more traditional techniques, such as mercury intrusion porosimetry, to generate a baseline for the technique. Further work will aim to understand which type of water is important in corrosion reactions and then determine how its mobility changes with time and ongoing hydration of the cement matrix.

### 4.13. Testing and assessment of historic samples

A programme of work to retrieve, test and assess the condition of inactive encapsulated wastes which have been in storage for over ten years has been recently completed. Samples which contain mild steel pucks of supercompacted waste and samples containing Magnox have been characterized. Very little corrosion has occurred to the metal in either type of sample. This can be seen by comparing the corrosion observed on a freshly encapsulated Magnox sample and a 13-year-old sample in Fig. 3.



Fig. 3 SEM of freshly encapsulated Magnox (left) and 13-year-old Magnox sample (right).

# 5. CONCLUSIONS

The corrosion of the metallic component of wastes is very important in understanding the short and long-term performance of waste packages. There are many complex issues, several of which are specific to the waste and encapsulating matrix, which need to be considered to gain the required understanding to allow successful encapsulation of metallic wastes.

The UK has over 20 years of experience in generating and interpreting the corrosion data required to support the encapsulation and disposal of metallic wastes in cement systems. A large amount of information on the corrosion of metallic wastes in these systems has been generated and the same principles and techniques can be applied to other encapsulation matrices.

### REFERENCES

- (1) INTERNATIONAL ATOMIC ENERGY AGENCY, Information Sheet for Coordinated Research Project on Disposal Aspects of Low and Intermediate Level Decommissioning Waste.
- (2) BUTCHER, E., et al, Development and implementation of technology for the treatment and encapsulation of operational intermediate level wastes, Icem 03-4735.
- (3) SETIADI, A., MILESTONE, N.B., HAYES, M., Corrosion of Aluminium in Composite Cements. Extended Abstract: Cement and Concrete Science, Warwick, 16-17 September 2004.
- (4) SETIADI, A., MILESTONE, N.B., HAYES, M., Corrosion of metals in BFS composite cements, Cement and Concrete Science, University of London 2005.
- (5) Patent application PCT/GB/2005/003376; Novel Encapsulation Medium.
- (6) ZHOU, Q., MILESTONE, N.B., HAYES, M., A new cement system for waste immobilisation - calcium sulfoaluminate cement system. In 'Stabilisation / solidification treatment and remediation advances in S/S for waste and contaminated land' A Al-Tabbaa, J A Stegemann. ISBN 04 1537 460 X.
- (7) BABOIAN, R., Corrosion tests and standards application and interpretation. ASTM manual series MNL 20. ISBN 0-8031-2058-3.
- (8) ASM Handbook Volume 13A Corrosion: fundamentals, testing and protection. ISBN 0-87170-705-5.

- (9) CALDWELL, R.J., RAWLINSON, S., BUTCHER, E.J., GODFREY, I.H., Characterisation of full – scale historic inactive cement based intermediate level nuclear wasteforms. In 'Stabilisation / solidification treatment and remediation advances in S/S for waste and contaminated land' A Al-Tabbaa, J A Stegemann. ISBN 04 1537 460 X.
- (10) Patent application PCT/GB2005/001423; Encapsulation of Hazardous Waste Materials.
- (11) HAYES, M., Improvement of grout performance for use in radioactive waste encapsulation processes. 'Powder characterisation: An industrial Perspective' Meeting of Royal Society of Chemistry - Particle Characterisation Interest Group (Analytical Division). Derby UK. November 2005.
- (12) http://www.shef.ac.uk/isl/

### UKRAINE

#### DECOMMISSIONING WASTE MANAGEMENT STRATEGY IN UKRAINE

#### O. Purtov, P. Rusinko, L. Litvinsky, A. Masko

State Scientific Engineering Center of Control Systems and Emergency Response, Kiev

#### Abstract

Optimization of decommissioning waste management scenarios for each NPP site in Ukraine is performed in this project. Optimization of the management of decommissioning of NPP units was proposed for the following parameters:

- Decommissioning waste generation;
- Time dependent expenditures for decommissioning;
- Number of personnel necessary for NPP decommissioning.

Prognoses of decommissioning radioactive waste volumes from NPP units with WWER reactor are calculated for each NPP in Ukraine. Expenses associated with transfer and disposal of decommissioning waste, and including necessary annual deductions corresponding to accumulation of costs for decommissioning, are estimated for each NPP site.

#### 1. INTRODUCTION

For the last ten years the Nuclear Energy Complex (NEC) has been an essential component of electricity power generation in Ukraine (more than 40%), making its continued operation an important requirement in the stable development of the national economy. The operation organization of all operating NPP units in Ukraine (Zaporozhye, Rovno, Khmelnitsky and South-Ukrainian NPP) is NNEGC "ENERGOATOM". Thirteen WWER-1000 reactor units and two units with WWER-440 reactors are in operation in Ukraine. Table 1 provides general information about operating NPP units in Ukraine.

NPP name No Unit Reactor type		Installed electricity capacity (MWt)	Start of Construct.	In grid connection	Design final shutdown	
Zaporozhye	1	WWER-1000/320	1000	04.1980	10.12.1984	10.12.2014
	2	WWER-1000/320	1000	04.1981	22.07.1985	22.07.2015
	3	WWER-1000/320	1000	04.1982	10.12.1986	10.12.2016
	4	WWER-1000/320	1000	01.1984	18.12.1987	18.12.2017
	5	WWER-1000/320	1000	07.1985	14.08.1989	14.08.2019
	6	WWER-1000/320	1000	06.1986	19.10.1995	19.10.2025
Rovno	1	WWER-440/213	415	08.1976	22.12.1980	22.12.2010
	2	WWER-440/213	420	10.1977	22.12.1981	22.12.2011
	3	WWER-1000/320	1000	02.1981	21.12.1986	21.12.2016
	4	WWER-1000/320	1000	1986	16.10.2004	10.10.2034
South Ukraine	1	WWER-1000/302	1000	03.1977	31.12.1982	31.12.2012
	2	WWER-1000/338	1000	10.1979	06.01.1985	06.01.2015
	3	WWER-1000/320	1000	02.1985	20.09.1989	20.09.2019
Khmelnitsky	1	WWER-1000/320	1000	11.1981	22.12.1987	22.12.2017
	2	WWER-1000/320	1000	1986	08.08.2004	07.08.2034

Table 1. General information for operating NPP units in Ukraine

The design life of WWER-440 and WWER -1000 reactor units is expected to be 30 years. Most operating NPP units have been in operation for more than half of their 30-year lifetime. Taking into account positive international experience with obtaining lifetime extensions for water-water reactors, the results of safety assessment analyses, and the realization of increasing safety level measures, Ukrainian nuclear authorities are considering prolonging the operation of the NPP units beyond their design life.

The basis for scheduling NPP decommissioning in Ukraine is provided in "Concept of decommissioning of operating nuclear power plants in Ukraine," approved in May 2004 [1] (Concept). The treatment of radioactive wastes generated during decommissioning is one of the essential components of the decommissioning process. The basic positions concerning decommissioning in the Concept (detailed in the given report) are based on optimized forecasts for arising decommissioning waste from NPP units in Ukraine, and costs necessary for transfer of decommissioning waste to a repository.

The Concept takes into account the ability of existing radioactive waste treatment and storage facilities to accept large quantities of waste that will arise during decommissioning. A strategy of decommissioning waste management is a key part of the Concept. Elaboration of this strategy allowed for the identification, in broad terms, of the requirements for the safe and efficient management of waste for planning purposes.

# 1.1. Purposes and Tasks

The main purposes and tasks of the project are as follows:

- Characterize and establish the inventory of large volume low-active waste and large components from decommissioning;
- Identification of quantitative estimates of all decommissioning waste streams arising at different stages of decommissioning;

- Analysis of the existing infrastructure for radioactive waste (RAW) management at the NPP sites and consideration of its use for decommissioning;
- --- Recommendations for elaboration and application of new waste management methods suitable for decommissioning waste;
- Substantiation of the construction of new installations and facilities needed for the provision of decommissioning;
- The following presently foreseen project stages:

Stage 1: Collection of information on the present conditions of the operating NPP units with the WWER type reactor,

- Stage 2: Collection of information on existing infrastructure for waste management and existing technologies for operational waste at NPP sites.
- Stage 3: Assessment of prospective volumes and streams of decommissioning waste at different stages of decommissioning.
- Stage 4: Optimization of the Strategy for decommissioning waste management in Ukraine.

### 2. TREATMENT OF DECOMMISSIONING WASTE FROM WWER REACTOR

#### 2.1. General scheme of waste treatment

The basic methods for radioactive waste treatment arisings are waste minimization, characterization and separation, reprocessing, conditioning, transportation and repository.

### 2.2. Treatment of solid decommissioning waste

The basic solutions concerning waste minimization and processing of solid radioactive waste (SRAW) should be identified during development of the programme (project) for decommissioning of operating units. It is planned to use already existing and planned NPP site utilities for treatment of operational and decommissioning waste.

Solid radioactive waste arising during decommissioning are also sorted by activity and, if necessary, fragmented. Solid radioactive wastes, depending on the level of contaminating, are classified into the three groups identified in Table 2. The estimation of no-reprocessing decommissioning waste generation for units with WWER-440 and WWER –1000 reactors are provided in Table 3 and Table 4, respectively.

Group RAW	Dose rate of γ-radiation at distance 0,1 м from surface SRAW	Specific β-activity, Ci/kg	Specific α - activity, Ci/kg	
SRAW I group	0,1 - 30 mR/hour	$2 \times 10^{-6} - 1 \times 10^{-4}$	$2 \times 10^{-7} - 1 \times 10^{-5}$	
(Low-activity)	(under 8 $\mu$ R/sec)	2×10 1×10	2~10 1~10	
SRAW II group	30-1000 mR/hour	$1 \times 10^{-4} - 1 \times 10^{-1}$	$1 \times 10^{-5} - 1 \times 10^{-2}$	
(Middle-activity)	(8-280 µR/sec)	1~10 - 1~10	1~10 - 1~10	
SRAW III group	1000 mR/hour and more	$1\times10^{-1}$ and more	$1 \times 10^{-2}$ and	
(High-activity)	(more 280 µR/sec)	1×10 and more	more	

 Table 2. Classification of solid radioactive waste in Ukraine

No	Decommissioning stage	Duration,	Production SRAW m <sup>3</sup> /year			
		years	SRAW - I	SRAW - II	SRAW -	
					III	
0	Termination of operation	4	84.41	1.41	0.67	
1	Final closure	5	201.96	5.05	0.50	
2	Preservation	4	25.24	18.93	1.89	
3	Safe storage	30	0.34	0.00	0.00	
4	Dismantling	9	112.20	11.22	11.22	

 Table 3. Estimation of non-reprocessible solid decommissioning waste generation for

 WWER-440 unit

# Table 4. Estimation of non-reprocessible solid decommissioning waste generation for WWER-1000 unit

No	Decommissioning stage	Duration,	Production SRAW m <sup>3</sup> /year			
		years	SRAW - I	SRAW - II	SRAW - III	
0	Termination of operation	4	201.12	3.96	0.31	
1	Final closure	5	302.93	7.57	0.76	
2	Preservation	4	37.87	28.40	2.84	
3	Safe storage	30	0.50	0.00	0.00	
4	Dismantling	9	168,30	16,83	16,83	

Low-activity waste is segregated according to its ability to be further processed. The following waste composition is assumed: burnable 30%, compressible 50%, metallic 10%, non-processible 10%. The current schedule utilizes the two basic waste processing technologies: super-pressing and burning. The waste volume minimization coefficients for super-pressing and burning are -5 and -10, respectively.

Recycling of metal waste via strong deactivation and/or re-melting will be considered. A final selection of technologies for conditioning solid radioactive waste is impossible, due to the absence of repository waste acceptance criteria in Ukraine. Possible technologies for waste conditioning include cementation, bitumination, vitrification, and polymerization. At the present time, the most likely technology for conditioning is cementation. For temporary storage of decommissioning waste, it is planned to use existing on-site capacities. The transfer of decommissioning waste to specialized disposal facilities is considered to be the final stage of decommissioning waste management.

# 2.2. Treatment with a liquid radioactive waste (LRAW)

It is assumed that decommissioning will generate several basic types of liquid radioactive waste: radioactive polluted water, waste tars (resins), sorbents, swafts, and radioactive - polluted oils. Currently, radioactive polluted water is evaporated. The estimated volume of evaporator bottoms generated at each decommissioning stage for reactors WWER - 440 and WWER -1000 are given in the Table 5.

No	Decommissioning stage	Duration,	Evaporator bottoms, m <sup>3</sup> /year	
		years	WWER - 440	WWER - 1000
0	Termination of operation	4	425.6	175.7
1	Final closure	5	73.3	175.7
2	Preservation	4	73.3	175.7
3	Safe storage	30	7.3	17.6
4	Dismantling	9	73.3	175.7

Table 5. Forecast of decommissioning LRAW volumes

The evaporator bottoms are assumed to be vaporized into salt melt (dry residual) in deep evaporation installations. The necessity of additional processing or conditioning of dry residual is under investigation. A solution concerning necessary technologies will be accepted once decommissioning waste acceptance criteria are developed based on repository requirements. The waste tars, sorbents, and swafts will require drying and containerization. Incineration of waste radioactive tars is currently being considered as the most likely LRAW processing technology. Radioactive polluted oils are planned to be incinerated together with combustible solid radioactive waste.

# 3. SCENARIOS OF DEVELOPMENT OF THE NEC OF UKRAINE

The Concept evaluates three Scenarios: prolongation of the 30-year operating life of NPP operating units by 10 years (Basic Scenario), by 10-15 years, and by 5-10 years, followed by decommissioning and deferred dismantling. Basic Scenario is presented in Figure 1.

4. RESULTS OF OPTIMIZATION OF DECOMMISSIONING SCENARIOES OF OPERATING NPPS IN UKRAINE

### 4.1. Bases for optimization and initial assumptions

Optimization was carried out for each NPP separately by varying the lifetime extension and terms of safe storage. Only the operating units were considered; the location of particular replacement units is not considered. The object of optimization was decommissioning Scenarios of operating units on NPP sites (below Variant - 0), which correspond to the basic Scenario of the Concept and provide operation term prolongation of all operating units for 10 years.

As in the Concept, unit labour costs and other costs associated with decommissioning waste are equivalent to those associated with operational waste. Total expenditures only vary as a result of the assumed durations and resulting volumes of waste arisings identified for each scenario.

The following basic parameters were analyzed during optimization: decommissioning waste generation; time dependent expenditures for decommissioning; and number of personnel necessary for NPP decommissioning. The optimized Scenarios of NPP lifecycles are designated as Variant 1.



Fig. 1 Basics scenario of NPP lifecycles (construction, operation and decommissioning).

# 4.2. Decommissioning waste management scenarios for Zaporozhye NPP

The optimized Scenario of Zaporozhye NPP decommissioning (Variant 1) was compared with the basic Scenario (Variant 0). As Figure 2 shows, Variant 1 allows significant reduction in the maximum arisings of decommissioning waste in comparison to Variant-0.

Annual generation of decommissioning waste: Only one waste volume peak is observed for Variant 1 at the termination of operation and preservation stages of Zaporozhye NPP units. However, this maximum, 571  $m^3/y$  in 2032, is lower in comparison with the maximum for Variant 0, which reaches 723  $m^3/y$ . The annual generation of waste from 2040 to 2063, approximately 280  $m^3/y$ , is uniform. The annual generation of waste during the period between 2064 and 2094 uniformly decreases from 147  $m^3/y$  to 133  $m^3/y$ . The sharp gap and peak appearing for Variant 0 in the year 2070 is missing for Variant 1. The analysis shows that Variant 1 ensures a more uniform use of waste processing utilities. A side benefit of Variant 1 in comparison with Variant 0 is more uniform execution of decommissioning operations, which will allow the utilities to fully exploit the accumulated experience and available decommissioning facilities, thus allowing for additional reduction of expenditures.

### 4.3. Decommissioning waste management scenarios for Rovno NPP

The optimized scenario of Rovno NPP decommissioning (Variant 1) was compared with the basic Scenario (Variant 0) using the above indicated parameters. As Figure 3 shows, generation of decommissioning waste for Variant 1 becomes more uniform, the peaks are flattened, and the sharp gaps of Variant 0 are missing.


Fig. 2 Prognoses of decommissioning waste generation rate for the Zaporozhye NPP.

Annual generation of decommissioning RAW: The forecasted annual generation of decommissioning waste for Rovno NPP operating units (Variant 1) has only one sharp peak in 2021 - 2023, with a maximum of  $273 \text{ m}^3/\text{y}$ . From the year 2030 to the end of decommissioning in 2085, the volumes of decommissioning waste generated annually for Variant 1 varies near 130 m<sup>3</sup>/yr, with a maximum deviation of 35 m<sup>3</sup>/y.

A side benefit of Variant 1 in comparison with Variant 0 is more uniform execution of decommissioning operations for operating units of Rovno NPP, which will allow the utilities to fully exploit the accumulated experience and available decommissioning facilities, thus allowing for additional reduction of expenditures.



Fig. 3 Prognoses of decommissioning waste generation for the Rovno NPP.

# 4.4. Decommissioning waste management scenarios for Khmelnitsky NPP

It was not possible to reach a complete optimization at the current stage of planning of Khmelnitsky NPP decommissioning. Results of comparison of the Variant 0 and the Variant 1 of the decommissioning waste management for Khmelnitsky NPP are presented on Figure 4.

As it can be seen, peak value changes are the same for both Variants, but for the Variant 1 there is no waste generation reduction between dismantling power units No 1 and No 2. This suggests personnel and equipment could be used more effectively for dismantling and waste treatment.

## 4.5. Decommissioning waste management scenarios for the South-Ukrainian NPP

Comparison of Variant 0 and Variant 1 of decommissioning waste management scenarios for SUNPP is presented on Figure 5; as can be seen, the optimization results in more uniform generation of waste.

Annual generation of the decommissioning RAW: At the termination of operation and preservation stages of SUNPP units the maximum annual volume of waste arisings decreases from about 420 m<sup>3</sup>/y to 300 m<sup>3</sup>/y in 2030. The maximal annual generation of decommissioning waste at dismantling stages decreases from 400 m<sup>3</sup>/y to 154 m<sup>3</sup>/y. The shape of annual decommissioning waste generation becomes more uniform, allowing the utilities to avoid equipment downtimes when processing waste.

A benefit of Variant 1 in comparison with Variant 0 is that stages of the termination of operation, final closure and preservation of power units No 2 and No 3 are carried out after the termination of corresponding stages at unit No 1. Similarly, at the stage of dismantling, all involved personnel can perform jobs serially on each of power units, until the stage of safe storage. Such organization of decommissioning activities will allow the utilities to fully exploit the accumulated experience and available decommissioning facilities, thus allowing for additional reduction of expenditures.



Fig. 4 Prognoses of decommissioning waste generation for Khmelnitsky NPP.



Fig. 5 Prognoses of decommissioning waste generation for South-Ukrainian NPP.

# 5. DYNAMICS OF THE GENERATION OF DECOMMISSIONING RAW OF OPERATING NUCLEAR POWER PLANTS IN UKRAINE

The forecast of the generation of decommissioning waste is shown in Figures 6 and 7 for the Scenario 0 and the Scenario 1 of NEC development, respectively. It is apparent from the figures that Scenario 1 (Optimized Strategy of the decommissioning waste management in Ukraine) provides more uniform waste generation that allows more optimum uses of waste treatment capacities, including transportation and transfer to a disposal facility. After selection of sites for constructing new power units, the additional optimization of Scenario 1 with corresponding decreases in the second peak of waste generation is possible.

## 6. DISPOSAL ASPECTS OF THE DECOMMISSIONING WASTE

Currently, the facilities of the Center for processing and disposal of so-called complex "VECTOR" in the Chernobyl Exclusive zone is supposed to be used for the disposal of short lived low- and intermediate activity level radioactive waste. This industrial complex for solid radioactive waste management is designed for removal, treatment and/or disposal of the accumulated operational and decommissioning wastes. Requirements for operational and decommissioning radioactive waste acceptance have not been declared.

## 7. CONCLUSIONS

Optimization of decommissioning scenarios of NPP power units for separate sites of NPP is performed in this project. The Optimization of the Strategy of the decommissioning waste management in Ukraine is proposed using the following parameters: decommissioning waste generation; time dependent expenditures for decommissioning; and number of personnel necessary for NPP decommissioning.

Prognoses of decommissioning waste arisings from WWER reactors units are calculated for each NPP site. Expenses for transport and disposal of decommissioning waste at a repository and the value of necessary annual deductions that correspond to uniform accumulation of costs for repository of decommissioning waste are estimated for each NPP.

#### REFERENCES

- [1] "Concept of decommissioning from operation of operating nuclear power plants in Ukraine", is approved by the order of Ministry of Fuel and Energy №249 from 5/12/2004.
- [2] The Law of Ukraine "About utilization of the nuclear energy and the radiating safety " from February, 8th 1995r. №39/95-Supreme Rada.
- [3] The Law of Ukraine "About the treatment with radioactive waste" from June, 30th, 1995 255/95-Supreme Rada.
- [4] The Reports on the project "Strategy of the decommissioning waste management in Ukraine" on IAEA Research Contract No: 12188/R1 RBF.
- [5] LITVINSKY, L., et al., Strategy of development of nuclear power in Ukraine up to 2030 and further perspective // Nuclear and radiation techniques, 2003, T.3, № 2, p. 74 [in Russian].
- [6] LITVINSKY, L., LOBACH, Yu., RUSINKO, P., Forecast of expenditures on WWER units decommissioning // Nuclear and radiation techniques, 2003, T.3, №3, p. 58 (in Russian).
- [7] RUSINKO, P., LITVINSKY, L., MASKO, A., Decommissioning Conception of the operating NPPs in Ukraine, The 10th International Conference on Environmental Remediation and Radioactive Waste Management, September 4-8, 2005, Glasgow, Scotland, ICEM05-1372.
- [8] INTERNATIONAL ATOMIC ENERGY AGENCY, Review of Selected Cost Drivers for Decision on Continued Operation of Older Nuclear Reactors (Safety Upgrades, Lifetime Extension, Decommissioning), IAEA-TECDOC-1084, IAEA, Vienna (1999).
- [9] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning Costs of WWER-440 Nuclear Power Plants, IAEA-TECDOC-1322, IAEA, Vienna (2002).

#### **UNITED STATES OF AMERICA**

#### CHALLENGES IN SOURCE TERM MODELLING OF DECONTAMINATION AND DECOMMISSIONING WASTES

#### T. Sullivan,

Brookhaven National Laboratory

#### Abstract

Understanding releases from decommissioning wastes, the source term, is essential for disposal of these wastes in a cost-effective manner that is protective of human health and the environment. Decommissioning wastes often include surface contaminated building materials, activated metals, and large pieces of equipment that differ from traditional low and intermediate-level wastes in their origin, radionuclide content, and physical and chemical form. In conducting a safety assessment the characteristics of D&D wastes need to be incorporated into the assessment. This paper discusses the differences between D&D wastes and routine operational radioactive wastes and provides a comparison of a source term analysis between operational and D&D waste streams.

#### 1. INTRODUCTION

Understanding releases from decommissioning wastes, the source term, is essential for disposal of these wastes in a cost-effective manner that is protective of human health and the environment. Decommissioning wastes arise from the termination of the use of facilities that have handled radioactive materials including nuclear power plants, submarines, and fuel processing plants. In addition, decommissioning wastes will arise from accelerators, nuclear medicine machines and facilities, and research institutions. Decommissioning wastes often include surface contaminated building materials, activated metals, and large pieces of equipment that differ from traditional low and intermediate-level wastes in their origin, and physical and chemical form.

Disposal of radioactive wastes requires a safety assessment that estimates potential health impacts of disposal to humans. Source term modelling predicts the rate of release of contaminants from a disposal facility and is an important process that needs to be quantified as a basis for a safety assessment. Source term modelling typically performed for low and intermediate level wastes needs to be specialized to address the unique properties of decommissioning wastes.

This paper will include an overview discussion of safety assessment to set the context for modelling of contaminant release from decontamination and decommissioning (D&D) wastes. The next section of the paper will discuss the sources of D&D wastes, the differences between these wastes and other more traditional wastes, and the challenges in modelling these wastes. An example of a new source term model for surface contaminated concrete wastes often found in D&D activities will be provided as a basis for understanding the modelling challenges for these wastes. Finally, conclusions on the challenges in source term assessment of D&D wastes will be provided.

## 2. THE SAFETY ASSESSMENT CONTEXT

Safety assessment has received substantial attention over the past decade. Guidance on how to conduct a safety assessment has been provided by the IAEA (IAEA, 1999, IAEA, 2003)

and others. A number of important concepts distinguish safety assessment for radioactive waste disposal from typical engineering analyses. These concepts lead to a definition that emphasize safety assessment as a multi-disciplinary, iterative process focused on regulatory compliance rather than an analysis of a disposal system for the purpose of predicting actual outcomes. Safety assessment involves site-specific, prospective evaluations of the post-closure phase of the system with three primary objectives: 1) to determine whether reasonable assurance of compliance with quantitative regulatory performance objectives can be demonstrated, 2) to identify data, design, and model development needs for reaching defensible decisions about regulatory compliance, and 3) to identify waste acceptance criteria related to quantities of wastes for disposal. This paper deals primarily with the first and second aspect in the definition.

Critical to the definition and objectives are the phrases *prospective modelling* and *reasonable* assurance. These phrases infer that the results are not intended to be interpreted as "predictions" of actual behaviour. The goal of safety assessment is to determine the conditions for which reasonable assurance of compliance with performance objectives can be provided; the goal is not to predict the actual outcome. Rather, the modelling is directed toward developing a sufficient understanding of the system behaviour to support decisions about design and closure conditions. Judgment will be a necessary part of the process of assessing the defensibility of the conceptual models because of the inherent uncertainties in the longterm processes and events. It is important to understand that the uncertainty discussed with regard to safety assessment is really the uncertainty with respect to the decision (i.e., regulatory compliance), not the uncertainty associated with the numerical results of the assessment. The safety assessment results are largely a function of the data, design, and assumptions considered in the analysis. Changes to any one of these can result in changes in the conclusions resulting from the assessment. To develop reasonable assurance, it is necessary to obtain an improved understanding of those aspects of system performance that are important to the decision; it will not require a perfect representation of all processes. This need for professional judgment requires that careful attention must be paid to documenting, justifying, and defending the conceptual models, data selections, and results.

Safety assessments require an analysis of the health impacts resulting form disposal of radioactive wastes. This is a complex problem involving many scientific disciplines. To make the problem tractable, the safety assessments are usually conducted by dividing the analysis into components. The major components are: (1) infiltration and cover performance, (2) waste container performance, (3) waste form performance, (4) transport through the vadose zone, (5) groundwater transport, (6) biotic transport, (7) atmospheric transport, and (8) exposure and health effects to man. The results from one component are used to provide input to the other components until potential exposure to radioactivity is assessed and compared to dose or risk-based regulatory standards.

Waste container and waste form performance can be crucial components in conducting a defensible safety assessment. Due to the potential for the waste container and waste form to control release over long periods of time, modelling of waste package and waste form performance is required. The next section discusses the origin of D&D wastes and how they differ from operational wastes. These differences create a need to understand their impact on container and waste form performance, items 2 and 3 in the above list of safety assessment components. An illustrative example of waste form performance for D&D wastes follows. The results of the model exercises are discussed in the context of how the modelling can help define data, design, and modelling needs to perform a defensible safety assessment.

#### 3. DECONTAMINATION AND DECOMMISSIONING WASTES

#### 3.1. Sources of D&D Wastes

At the end of their useful life nuclear facilities such as nuclear power plants, nuclear submarines, reprocessing facilities, accelerators, nuclear medical facilities, and research facilities are decontaminated and decommissioned. The type of radioactive contamination will depend on the function of the facility. For example, the major source of contamination in an accelerator facility is likely to be in the form of activated metals and concrete, whereas the principle concern in a fuel processing facility would probably be surface contamination. Some of the contamination will be in solid forms (e.g., metal structures) that can be handled relatively easily. However, other contamination may be in the form of powder or dust, and loosely attached to structures. In most cases, the identities and forms of the major contaminants can be deduced from the operational histories of these facilities but their actual magnitudes and distribution throughout those facilities must be determined through characterization.

Decontamination is the removal of surface or near-surface contamination from a component or facility. This contamination, in the case of nuclear facilities, is predominantly of the radioactive type. However, other hazardous (but non-radioactive) contamination may be present; for example, surfaces in older buildings may be covered in lead-based paints and piping insulated with asbestos.

There are several reasons why decontamination is beneficial in conducting decommissioning programmes but sometimes the advantages may be far outweighed by other factors such as cost and total dose. Decontamination will not reduce the total radioactivity associated with a facility but only redistribute it. By reducing the activity associated with one particular area or component, subsequent work performed there may be done more efficiently and with less exposure for the workers. However, the radioactivity so removed then enters the secondary waste stream where it will need to be handled and processed before final disposition. These wastes will be treated similarly to existing operational wastes and are not a focus of this paper.

Upon completion of decontamination, dismantlement occurs. Dismantlement of a nuclear facility and, if required, its complete demolition can be accomplished using a mixture of old established methods and newer technologies recently developed to address specific problems. The methods available for dismantling metal components and structures usually fall into one of two categories: mechanical (saws, cutters, wrenches, etc.) and thermal (flame cutting, plasma arc cutting, etc.). Concrete and other non-metallic structures can be taken apart using equipment such as the abrasive water jet. When the dismantlement involves simple disassembly or cutting apart of non-contaminated equipment then generally conventional established methods will suffice. However, robot work systems equipped with appropriate tools are often deployed when the dismantling must be done in a radiation field. The resulting wastes are much different than operational wastes and often are comprised of large metallic or concrete components.

## **3.2.** Major differences between D&D and other Wastes

Major differences in contaminated materials resulting from decommissioning projects include:

- Size decommissioning wastes often contain large pieces of contaminated equipment and structures. These large pieces may invalidate the assumption of a uniform distribution of wastes within a disposal cell that is frequently used in Safety Assessment.
- Physical and chemical characteristics decommissioning wastes contain construction debris (concrete, wood, steel) and equipment components.
- Distributions of contaminants decommissioning wastes are often surface contaminated while low-level wastes are treated to form a more homogeneous waste form. This non-uniform distribution makes estimation of the total inventory difficult in many cases. Safety analyses typically assume a uniform concentration within the waste form.
- Limited data little direct data exists on the release of contaminants from decommissioning wastes. Data may not be sufficient for safety analysis (for example, measures of gross alpha or beta are often taken for making decisions on worker protection, but these are inadequate for safety assessment where the inventory on a radionuclide specific basis is required.

# **3.3.** Source Term Modelling

Source term modelling involves predicting the release and transport of contaminants out of the disposal facility. This involves understanding of barrier performance (concrete vaults, backfill, etc.), infiltration, container performance, waste form performance (leach rate), and transport out of the disposal facility. D&D wastes are not expected to have a major impact on infiltration, barrier performance, or transport processes after release. Therefore, the major differences between D&D wastes and more traditional wastes are expected in the waste form performance and possibly, to a lesser degree, t container performance. Differences in waste form performance are the focus of this paper.

Based on the expected physical and chemical processes for release, a number of general modelling approaches to simulate the release of contaminants from waste materials have been developed. These can be, perhaps with modification, applied to D&D wastes and include:

- Instantaneous release of the entire inventory. This is the most conservative case and can be used to simulate a surface rinse process. If projected doses are less than regulatory limits no further characterization is necessary.
- Sorption release. This model assumes that the entire inventory is available for release; however, there is an equilibrium sorption relationship that controls release. In this case, the release rate is controlled by the flow of water around the waste forms. Higher water flow provides more water per unit time to remove contaminants.
- Solubility limited release. Releases of some radionuclides are controlled by their solubility which depends on the chemical conditions near the waste and may be difficult to estimate. Again, the water flow rate controls the release rate.
- Diffusion release. Many cement materials exhibit diffusion controlled release. In this case, the waste controls release and release is independent of flow.
- Dissolution release. This model represents chemical reactions that dissolve the surface of a material (e.g. corrosion). This is a complicated process depending on the material, chemistry (pH, Eh, dissolved ions, etc.) and the environment (temperature, relative humidity in unsaturated systems, etc.). The simplest approach to address this is to define a constant dissolution rate. This may be the most appropriate model for activated equipment components.

## 3.4. Challenges in Modelling D&D Wastes

Safety Assessment typically assumes that the wastes can be homogenized over the disposal facility. This is often a reasonable approximation for facilities that treat all of their wastes using the same process (e.g. cementation). The D&D wastes challenge these assumptions due to their differences from operational wastes as discussed in Section 3.2.

In particular, D&D wastes may have large equipment components that are activated metals, large blocks of concrete that are either surface contaminated or have non-uniform concentrations with largest values near the surface and decreasing with depth into the concrete, and they will have materials not normally found in operational wastes. These inhomogeneities may also challenge assumptions used to assess intruder scenarios. Finally, there is often very little direct data on leaching characteristics of these wastes. For example, general corrosion rates of metals may be available, but leaching of trace impurities may not be available. Therefore, assumptions that the trace impurities are released at the same rate as the dissolution of the metal can not be proved or disproved.

# 4. ILLUSTRATIVE EXAMPLE: MODELLING RELEASES FROM D&D WASTE STREAM

Modelling can be used as a tool to determine the impacts of the differences in D&D wastes on long term safety. A literature review identified several waste streams generated during decommissioning of nuclear facilities. For illustrative purposes, a representative waste stream was selected for source term analysis (1). Table 1 presents a comparison of the radionuclide inventory of the operational wastes at the Saratov site (2) and the D&D waste stream at Hanford. The D&D waste stream was analyzed for 45 different radionuclides with concentrations ranging from 5.6 Ci/m<sup>3</sup> for Co-60 to non-detects for many radionuclides. Based on this data and preliminary screening calculations, a safety assessment analysis was performed for the eight radionuclides in Table 1. The nuclide distribution in Table 1 is much different than for typical processing wastes which are frequently dominated by Cs, Co, and Sr. In addition, the D&D wastes contain some nuclides not found in the Saratov wastes and that are mobile in the environment (Tc-99 and Cl-36).

Radionuclide	Operational Wastes		D&D Wastes	
	Ci	Ci/m <sup>3</sup>	Ci/m <sup>3</sup>	
Cs-137	6.23E+02	3.12	1.18	
Co-60	1.03E+02	0.51	5.6	
H-3	2.48E+01	0.124	0.19	
Pu-Be	2.21E+01	1.11E-01	-	
Pu-239	5.66E+00	2.83E-02	-	
Am-241	3.57E+00	1.79E-02	-	
Sr-Y	7.37E-01	3.69E-03	-	
T1-204	3.94E-01	1.97E-03	-	
Th-232	7.50E-02	3.75E-04	-	
Pm-147	2.34E-02	1.17E-04	-	
Kr-85	8.46E-03	4.23E-05	-	
C-14	2.00E-03	1.00E-05	2.6E-03	
Ni-63	-	-	1.1	
Nb-94	-	-	2.6E-04	
C1-36	-	-	5.6E-05	
Tc-99	-	-	2.8E-05	
Total	7.83E+02	3.92E+00		

# Table 1. Comparison of operational wastes at the Saratov site in200 m³ and D&D wastes

This project investigated several different scenarios based on the amount of credit claimed for the waste container and waste form. All analyses were performed using the DUST-MS computer code (3). The base case assumed no credit for the container and waste form. The inventory is released immediately after emplacement. The second case assumes that the container lasts for 100 years followed by release of the total inventory at that time. The third case is based on the assumption that after the container fails, release is controlled to 1% of the inventory per year. This is meant to represent the slow corrosion of metallic components which comprised the bulk of this waste stream.

The conceptual model of the facility is presented in Figure 1. The physical dimensions were selected to match the disposal vaults at the Saratov site. The waste zone is 2.7 m thick and 15.75 meters wide. The total volume is  $200 \text{ m}^3$ . The receptor well is 100 m down gradient from the disposal site. The vadose zone is 10 m and the aquifer is 5 m thick.



## Fig. 1 Conceptual model of disposal site.

Table 2 contains the U.S. Drinking water standards, distribution coefficient, and half-life of the more mobile radionuclides in the D&D wastes. The drinking water standards are used as a basis for comparison to avoid the need to perform a complete dose assessment.

Table 3 presents the material properties and Darcy flow rate for each region during the simulation. Again, these values are based on the Saratov site. For the vault region, flow is 1 percent of infiltration for the first 100 years, representing an intact vault system. After that time, the value increases to the infiltration rate of 35 cm/yr.

Isotope	Drinking water standards (pCi/L)	Distribution coefficient, K <sub>d</sub> (cm <sup>3</sup> /g)	Half life (y)
H-3	20,000	0.0	12.33
C-14	2,000	5.0	5730
Cl-36	2,000	0.8	301,000
Tc-99	900	0.1	213,000

## Table 2. Radionuclide properties

Zone	Material	Density	Moisture	Dispersivity	Darcy	Velocity
		$(g/cm^3)$	content	coefficient	(cm/yr)	
				(cm)	Before	After
					100 y	100 y
	Sand	1.5	0.20	30	0.35	35
	(Cap)					
	Concrete	2.2	0.15	30	0.35	35
	(Cover)					
Vault	Sand	1.5	0.20	30	0.35	35
	Waste	1.2	0.05	30	0.35	35
	zone					
	Concrete	2.2	0.15	30	0.35	35
	(Bottom)					
Vadose	Soil	1.9	0.05	100	35	35
Zone (VZ)						
Saturated	Soil	1.5	0.40	100	292	292
Zone (SZ)						

Table 3. Material Properties for transport

## 4.1. Model Results

The base case for the model results uses the assumption that all containers fail instantly and all radionuclides are released instantly. With these assumptions none of the radionuclides in the operational inventory exceeded the drinking water standards. However, H-3, C-14, and Tc-99 exceeded the drinking water standards in Table 2 and Cl-36 was close to the standard. Co-60, Cs-137, Ni-63, and Nb-93 all reached the receptor well at concentrations less than 1 pCi/l well below the drinking water standard for each radionuclide. The reason for this is due primarily due to the high distribution coefficient for each of these (Kd = 60, 10000, 400 and 160 for Co, Cs, Ni, and Nb, respectively) and for most of these the short half-life.

Figure 2 contains the predicted concentrations for H-3, C-14, Cl-36, and Tc-99 for the three simulated conditions, instant container failure and release, container failure at 100 years with instant release, and container failure at 100 years with release of 1% per year. H-3 is the only radionuclide that shows an effect of the 100 year container lifetime. Prior to 100 years, the flow is controlled by the engineered barriers and most of the movement occurs after



Fig. 2 Concentration profiles over time for H-3, C-14, Cl-36, and Tc-99.

100 years when water flow through the facility is at the infiltration rate. For H-3 due to its short half life, the impact of early failure is visible. In this example, it takes the concentration from slightly above the drinking water standard for instantaneous failure to just below the standard for container failure at 100 years. For the long-lived radionuclides, container failure at 0 or 100 years did not have a meaningful impact on concentration at the receptor well. In this illustrative example, C-14, and Tc-99 are above the drinking water standard for instantaneous release after container failure. Using a waste form release rate of 1% per year, lowers the peak concentration of Tc-99 by more than a factor of 5 and the peak concentration is lower than the drinking water standard. For C-14, the impact of the slow release from the waste form was minimal (< 10%) for peak concentration. This is because of sorption which causes the transport time to the receptor to be far in excess of the duration of release, (100 years). In this case, the peak concentration is controlled by the transport characteristics and not the disposal facility characteristics.

#### 5. CONCLUSIONS

D&D wastes present some unique challenges in safety assessment. Their large size, nonhomogeneity of contaminants, atypical, as compared to other wastes, material properties, and absence of long-term data on their ability to retain radionuclides distinguish them from the common waste types. Safety assessment provides the framework for evaluating the impacts of these differences on the ability to demonstrate that the wastes can be safely disposed. However, typical assumptions pertaining to heterogeneity of the wastes and disposal facility are severely challenged by D&D wastes. The major differences in the Safety Assessment for D&D wastes and more typical operating wastes primarily deal with the release of radionuclides from the waste form, a major component of source term modelling. Within the framework of a safety assessment, source term modelling can be specialized to examine the impact of the unique characteristics of D&D wastes. The results of these analyses can be used to determine impacts on safety and define characterization needs.

An illustrative example comparing peak groundwater concentrations from operational wastes and D&D wastes was performed. In the example, none of the radionuclides from the operational waste caused groundwater concentrations to exceed drinking water standards even when credit was not given for the container or waste form. In contrast, three radionuclides, H-3, C-14, and Tc-99 were in excess of the drinking water standard based on the D&D inventory levels. Taking credit for 100 year container life time reduced peak H-3 concentrations to below the drinking water standard. Taking credit for the container and a waste form release rate of 1% per year, Tc-99 was below the drinking water standard. C-14 remained above the standard for all simulations.

The example provided was for illustrative purposes only. It is not meant to represent any particular site or waste stream. The point of the example was that D&D wastes are different than operational wastes and will require different data and analysis in Safety Assessments.

#### REFERENCES

- [1] Fluor Hanford, "Solid Waste Integrated Technical Forecast (SWIFT) 2004 2035," HNF-EP-0918, Rev 12, Vol. 12, 2004.
- [2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Near Surface Disposal of Radioactive Waste: Safety Guide, Safety Standards Series No. WS-G-1.1, IAEA, Vienna (1999)
- [3] INTERNATIONAL ATOMIC ENERGY AGENCY, Co-ordinated Programme on Improvements of Safety Assessment Methodologies for Near Surface Disposal Facilities (ISAM), Volume I Review and enhancement of safety assessment approaches and tools, IAEA, Vienna (2003).
- [4] SULLIVAN, T.M., "DUSTMS\_D Disposal Unit Source Term Multiple Species Distributed Failure Data Input Guide. Rev 1.," BNL-75554-2006, Brookhaven National Laboratory, Upton, NY, 11973, January, 2006.

# CONTRIBUTORS TO DRAFTING AND REVIEW

Ali, S.S.	NRGP (K), India	
Borrmann, F.	Bundesamt für Strahlenschutz (BfS), Germany	
Carlsson, J.	Nuclear Fuel and Waste Management Company, Sweden	
Dayal, R.	International Atomic Energy Agency	
Godfrey, H.	British Nuclear Fuels Ltd, United Kingdom	
Hanušik, V.	VUJE, Trnava, Slovakia	
Harriague, S.	Comisión Nacional de Energía Atómica, Argentina	
Husain, A.	Kinectrics Inc. Canada	
Kim, C-L.	KHNP, Republic of Korea	
Koryakovski, Y.	St. Petersburg State Institute of Technology, Russian Federation	
Kumar, S.	Bhabha Atomic Research Centre, India	
Lobach, U.	State Scientific Engineering Centre of Control Systems and Emergency Response, Ukraine	
Nachmilner, L.	International Atomic Energy Agency	
Nechaev, A.	St. Petersburg State Institute of Technology, Russian Federation	
Poskas, P.	Lithuanian Energy Institute, Lithuania	
Purtov, O.	State Scientific Engineering Centre of Control Systems and Emergency Response, Ukraine	
Sullivan, T.	Brookhaven National Laboratory, United States of America	
Takáts, F.	TS Enercon Kft., Hungary	
Warnecke, E.	Bundesamt für Strahlenschutz (BfS), Germany	
Wen, Z.	Beijing Research Institute of Uranium Geology, China	

## **Research Coordination Meetings:**

Sellafield, United Kingdom: 10–14 March 2003 Buenos Aires, Argentina: 13–17 September 2004 Vienna, Austria: 20–24 February 2006