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In cooperation with the OECD Nuclear Energy Agency

Hosted by the Government of Japan through the Nuclear and Industrial Safety Agency (NISA), Ministry of Economy, Trade and Industry (METI), in cooperation with the Japan Nuclear Energy Safety Organization (JNES)

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IAEA - NEA

International Conference
on the
Safety of Radioactive Waste Disposal

Tokyo, Japan
3-7 October 2005

CONTRIBUTED PAPERS

IAEA-CN-135
NOTE

The International Atomic Energy Agency (IAEA), in co-operation with the Nuclear Energy Agency of the Organization for Economic Co-operation and Development (OECD/NEA), is organizing an International Conference on the Safety of Radioactive Waste Disposal, to be held at Tokyo, Japan, from 3 to 7 October 2005. The Government of Japan is hosting this Conference and is facilitating its organization through the Nuclear and Industrial Safety Agency (NISA), the Ministry of Economy, Trade and Industry (METI), in co-operation with the Japan Nuclear Energy Safety Organization (JNES).

This book contains concise technical papers and posters contributed on issues falling within the thematic scope of the Conference which were accepted following the guidelines established by the Conference Programme Committee for consideration at the Conference. The material compiled in this book has not undergone rigorous editing by the editorial staff of the IAEA. However, certain modifications were made: a unified format was adopted for all papers, abstracts were added when missing, and minor corrections were made in the text where required. The IAEA wishes to express appreciation to all authors for their contributions made to the Conference.

It is intended that, after the Conference, the contents of this book will be published in form of a CD ROM as part of the proceedings of the Conference. Authors wishing to make slight modifications or corrections to their paper are encouraged to contact the Conference Secretariat.

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Safety assessment for the Chinese Beilong LILW disposal facility using AMBER

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People's Republic of China

Abstract. The Beilong disposal site is a planned national regional Low and Intermediate Level Waste (LILW) disposal facility located near Daya Bay NPPs of Guangdong Province in South China. This paper describes the implementation of a conceptual model for the Beilong LILW disposal site into the AMBER software tool. The features of the facility are provided and the conceptual model development is given with application of AMBER for the safety assessment.

1. Introduction

The development of a disposal facility passes through several distinct phases including site selection, facility construction, disposal operations and final site closure. At each stage, a safety assessment is often used to support decisions. The purpose is to assess the level of safety of the disposal facility using currently available information. The assessment of the safety of radioactive waste disposal in near surface formations is a multi-step process involving the specification of the assessment context; the description of the disposal system; the development and justification of scenarios; the formulation and implementation of models; and the presentation and analysis of results.

The model formulation and implementation step involves taking information from the assessment context, system description and scenario development steps of the safety assessment to help generating conceptual models of the disposal system. These conceptual models and their associated processes are represented in mathematical models that are then implemented in computer codes. Throughout this process, data are used to help developing the conceptual and mathematical models and input to the computer codes.

AMBER is a flexible software tool that allows users to build their own dynamic compartment models through the use of a graphical user interface to represent the migration and fate of contaminants in a system. It has been used to model routine, accidental and long-term releases of radioactive contaminants in solid, liquid and gaseous phases.

The Beilong disposal site is a planned national regional Low and Intermediate Level Waste (LILW) disposal facility located near Daya Bay NPPs of Guangdong Province in South China. The designed total capacity is 80 000 m$^3$ with 70 cells with radioactivity $5.4\times10^{15}$ Bq. The first construction phase began in 1998 and was completed in 2000. At present, the facility is not in operation and is awaiting the license.

This paper focuses on the implementation of conceptual model for Beilong LILW disposal site into the AMBER software tool. Firstly the features of the facility are provided and then the conceptual model development is given, with application of AMBER for the safety assessment.

2. Scenario description

The facility is located in a tropical climatic zone with an annual average precipitation of 1900mm and an annual average evaporation of 1600mm. The area around the disposal site on Dapeng Peninsula is hilly. The waste was fixed by cement. The concrete cover is built as engineering barrier. Fig 1 shows a cross section of the disposal facility and associated engineered barriers.
The waste is obtained from Daya Bay NPPs. Based on operating experience, $^{137}$Cs, $^{60}$Co, $^{90}$Sr, $^{63}$Ni, $^{14}$C, $^{239}$Pu, $^{3}$H are selected as the critical radionuclides of long term impact to the environment at the facility closure. Table 1 shows the radionuclides inventory.

Table I. Radionuclides inventory at closure

<table>
<thead>
<tr>
<th>Radionuclide</th>
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<tr>
<td>$^{137}$Cs</td>
<td>1.1E+15</td>
</tr>
<tr>
<td>$^{60}$Co</td>
<td>3.3E+15</td>
</tr>
<tr>
<td>$^{90}$Sr</td>
<td>3.7E+11</td>
</tr>
<tr>
<td>$^{63}$Ni</td>
<td>9.6E+14</td>
</tr>
<tr>
<td>$^{14}$C</td>
<td>3.3E+13</td>
</tr>
<tr>
<td>$^{239}$Pu</td>
<td>3.4E+11</td>
</tr>
<tr>
<td>$^{3}$H</td>
<td>5.4E+15</td>
</tr>
</tbody>
</table>

3. Development of conceptual model

The radionuclides transfer pathway is shown in Fig 2. The scenario essentially considers the main exposure pathway to be the leaching of the waste by rainwater infiltrating through the cap, followed by the subsequent migration of radionuclides through the geosphere to the groundwater used by humans in drinking and seafood ingestion. Therefore, some compartments are constructed in the conceptual model which are disposal facility, unsaturated zone, aquifer, estuaries, and human.

Application of AMBER for the safety assessment

The conceptual model forms the basis for the development of a mathematical model that is then implemented in the computer code used for the assessment calculation. AMBER can be applied in the compartmental modelling approach associated with data and equation to represent the migration and fate of contaminants in this scenario. Fig 3 shows the safety assessment implementation of the conceptual model in geosphere and biosphere by AMBER software.
The mathematical calculation considers the cement solid, concrete container, backfill material, treatment unit, unsaturated zone, aquifer, river and spring water. It is assumed that the radionuclides are uniformly distributed.

The dose resulting from drinking water is from \( D = V \sum C_i F_i \exp (-\lambda_i t) \), where \( D \) is the dose from drinking contaminated water, Sv/yr; \( V \) is the drinking amount of each year in m\(^3\), in which the value is 0.4 for infant, 0.5 for teenager, and 0.73 for adult; \( C_i \) is the concentration of radionuclide in water in Bq/m\(^3\); \( F \) is the ingestion dose factor in Sv/Bq.

The dose resulting from marine food ingestion is from \( D = qF_i F M \), where \( F \) is the ingestion dose factor in Sv/Bq. \( F_i \) is the concentration factor of marine food to radionuclides in m\(^3\)/kg. \( M \) is the ingestion amount of marine food each year, in which the value for adult is 15 kg of fish, 5 kg of...
crustacean, and 1.5 kg of molluscs; the value for teenager is 10 kg of fish, 3.5 kg of crustacean, and 0.9 kg of mollusks; the value for infant is 8 kg of fish, 3 kg of crustacean, and 0.7 kg of molluscs. C is the concentration of radionuclides in seawater in Bq/m³, q is the annual radionuclides release rate to the sea, 1/y.

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Mathematical interpretation of transport phenomena in porous materials

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Abstract. The leaching rate of $^{137}$Cs from spent mix bead (anion and cation) exchange resins in a cement matrix has been studied. Transport phenomena involved in the leaching of a radioactive material from a cement composite matrix are investigated using three methods based on theoretical equations. These are: the diffusion equation for a plane source, an equation for diffusion coupled to a first-order equation, and an empirical method employing a polynomial equation. The results presented in this paper are from a 25-year mortar and concrete testing project that will influence the design choices for radioactive waste packaging for a future Serbian radioactive waste disposal centre.

1. Introduction

Ion exchange resins may be used most successfully for the removal of radioactive and stable ions from dilute solutions. Ion exchange resins are polymers with cross-linking (connections between long carbon chains in a polymer). The resin has active groups in the form of electrically charged sites. At these sites ions of opposite charge are attached but may be replaced by other ions depending on their relative concentrations and affinities for the sites. Spent mix bead exchange resins containing $^{60}$Co and $^{137}$Cs represent a major portion of the solid radioactive waste in nuclear technology.

Cement is used as a solidification material for the storage of intermediate-level radioactive waste. However, the retention of caesium, in the cement matrix is negligible. The sorption of caesium on cement is low and diffusion of caesium in the hydrated cement is high. Only when the cement is mixed with a material having a significant sorption capacity, normally bead or powdered ion exchange resins, is the leaching of caesium and cobalt from the cement matrix low enough to be acceptable.

At the "Vinca" Institute of Nuclear Sciences, a promising composite for the solidification of radioactive wastes has been developed. Leaching of $^{137}$Cs from this material was studied using the method recommended by the IAEA [1].

2. Theoretical methods

Below, three methods are compared with respect to their applicability to experimental leaching data [2,3,4,5,6,7,8,9,10,11,12,13].

2.1. Method I: diffusion equation based on a plane source model

In this model the fraction $f$ leached at time $t$ (d) is given by [1]

$$f = \frac{\sum a_n}{A_o} = 2 \frac{S \sqrt{D_e} t_n}{V \sqrt{\pi}}$$

(1)

where $\Sigma a_n$ is the cumulative fraction leached of contaminant for each leaching period, $A_o$ is the initial amount of contaminant in the sample, $V$ is the volume of sample (cm$^3$), $S$ is the exposed surface area of the sample (cm$^2$), $t_n$ the duration of leachant renewal period (d) and $D_e$ is the diffusion coefficient (cm$^2$ d$^{-1}$).

The results may also be expressed by the cumulative fraction of the contaminant. Leach test results are plotted as the cumulative fraction of contaminant leached from the samples as a function of the square root of total leaching time.
If the model is applicable a plot of $\frac{\Sigma a_n}{A_o}$ versus $\sqrt{\Sigma t_n}$ is a straight line and the diffusion coefficient $D_e$ is given by:

$$D_e = \frac{\pi}{4} m^2 \frac{V^2}{S^2}$$ (3)

where $m=(\Sigma a_n/A_o) (1/\sqrt{\Sigma t})$, is the slope of the straight line ($d^{-1/2}$).

2.2. Method II: rate equation for coupled diffusion and simultaneous first-order reaction

In this model, the rate equation is

$$\frac{\partial C}{\partial t} = D_e \left( \frac{\partial^2 C}{\partial X^2} \right) + g(C)$$ (4)

Here, the special case where $g(C)$ is directly proportional to the concentration $C$, i.e. a first-order reaction was considered. The initial and boundary conditions are,

$$t = 0, \ x > 0, \ C = C_o$$ (5)
$$t = 0, \ x < 0, \ C = 0$$ (6)
$$t > 0, \ x = 0, \ C = 0$$ (7)

From this, the fraction leached from a specimen having a surface area $S$(cm$^2$) and volume $V$(cm$^3$) is

$$f = (S/V) \sqrt{D_e/k} \left[ \left( kt + 1/2 \right) \text{erf} \left( \sqrt{kt} + \sqrt{kt/\pi} \exp(-kt) \right) \right]$$ (8)

Where $k$ is the rate constant of the first-order reaction and

$$\text{erf}(u) = \frac{2}{\sqrt{\pi}} \int_{0}^{u} \exp(-z^2) \, dz$$ (9)

2.3. Method III: polynomial equation

The orthogonal polynomial is one of the most useful empirical equations. Its general form is:

$$y(x) = \sum_{i=1}^{n} A_i \phi_i(x)$$ (10)

where $A_i$ is the parameter to be determined, and $\phi_i$ is a function of $x$.

Here, $\phi_i(x)$ - is taken as $t^{1/2}$, and the leaching fraction is given by

$$f = \sum_{i=1}^{n} A_i t^{1/2}$$ (11)

To simplify the mathematical treatment, a five terms polynomial of the form

$$f = A_0 + A_1 t^{1/2} + A_2 t + A_3 t^{3/2} + A_4 t^2$$ (12)

was fitted to the leaching data.
For this type of model, extrapolation to longer term leaching is not advisable since the arbitrary constants do not necessarily have any physical significance.

3. Preparation of sample for leaching test

The grout samples were prepared from a standard Portland cement. The cement was mixed with saturated wet non-radioactive mix bead exchange resins (Lewatit S 100), additive-bentonite clay (63% SiO₂; 18% Al₂O₃; 4% Fe₂O₃; 2.6% MgO and 3.3% CaO), and water with artificial radioactivity of ^{137}Cs, A₀ = 60(kBq), in the reson to simulate radioactive spent ion exchange resins. Mixing time was about ten minutes. The mixtures were cast into 50 mm diameter cylindrical molds with a height of 50 mm, which were then sealed and cured for 28 days prior to the leaching experiments. Leaching of ^{137}Cs was studied using the method recommended by the IAEA, [1]. The duration of leachant renewal period was 30 days. More then 100 different formulations of grout form were examined to optimize their mechanical and sorption properties. In this paper, we discuss four representative formulations. Grout composition formulas are shown in Table I.

Table I. Grout compositions (calculated as grams for 1000 cm³ of samples)

<table>
<thead>
<tr>
<th>Materials (g)</th>
<th>G₁</th>
<th>G₂</th>
<th>G₃</th>
<th>G₄</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mix bead ion exchange resins</td>
<td>324</td>
<td>327</td>
<td>331</td>
<td>341</td>
</tr>
<tr>
<td>Portland cement</td>
<td>1337</td>
<td>1318</td>
<td>1300</td>
<td>1290</td>
</tr>
<tr>
<td>Water with artificial radioactivity of</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>^{137}Cs, A₀ = 60(kBq)</td>
<td>277</td>
<td>269</td>
<td>260</td>
<td>250</td>
</tr>
<tr>
<td>Bentonite</td>
<td>13</td>
<td>13</td>
<td>13</td>
<td>13</td>
</tr>
</tbody>
</table>

4. Results

Experimental data show the fractions of ^{137}Cs leached from grout composite as a function of the square root of the leaching period. The linear relation between f and t is not observed throughout the test period. From the application of Method I to the leaching data we obtained:

\[ f(G₁) = 4.3 \times 10^{-4} t^{1/2} + 6.4 \times 10^{-9} \]
\[ f(G₂) = 4.8 \times 10^{-4} t^{1/2} + 6.9 \times 10^{-9} \]
\[ f(G₃) = 5.2 \times 10^{-4} t^{1/2} + 7.9 \times 10^{-9} \]
\[ f(G₄) = 6.6 \times 10^{-4} t^{1/2} + 9.2 \times 10^{-9} \]

The diffusion coefficients predicted by Method I are:

\[ D₁(G₁) = 4.2 \times 10^{-6} \text{ (cm}^2/\text{d}) \]
\[ D₁(G₂) = 4.8 \times 10^{-6} \text{ (cm}^2/\text{d}) \]
\[ D₁(G₃) = 7.1 \times 10^{-6} \text{ (cm}^2/\text{d}) \]
\[ D₁(G₄) = 8.7 \times 10^{-6} \text{ (cm}^2/\text{d}) \]

Method II was applied to the leaching data to obtain the unknown parameters D₂ and k. From this we obtained:

\[ D₂(G₁) = 2.70 \times 10^{-6} \text{ (cm}^2/\text{d}) \]
\[ D₂(G₂) = 3.90 \times 10^{-6} \text{ (cm}^2/\text{d}) \]
\[ D₂(G₃) = 4.3 \times 10^{-6} \text{ (cm}^2/\text{d}) \]
\[ D₂(G₄) = 6.40 \times 10^{-6} \text{ (cm}^2/\text{d}) \]

Using the least squares procedure, Method III yielded:

\[ f_{III}(G₁) = 4.4 \times 10^{-8} + 3.8 \times 10^{-4} t^{1/2} + 3.2 \times 10^{-8} t + 7.4 \times 10^{-12} t^{3/2} + 4.8 \times 10^{-6} t^2 \]
\[ f_{III}(G₂) = 3.7 \times 10^{-8} + 5.9 \times 10^{-4} t^{1/2} + 4.5 \times 10^{-8} t + 8.1 \times 10^{-12} t^{3/2} + 5.8 \times 10^{-6} t^2 \]
\[ f_{III}(G₃) = 3.0 \times 10^{-8} + 6.6 \times 10^{-4} t^{1/2} + 5.4 \times 10^{-8} t + 9.2 \times 10^{-12} t^{3/2} + 6.6 \times 10^{-6} t^2 \]
\[ f_{III}(G₄) = 2.4 \times 10^{-8} + 8.4 \times 10^{-4} t^{1/2} + 7.6 \times 10^{-8} t + 9.9 \times 10^{-12} t^{3/2} + 7.4 \times 10^{-6} t^2 \]
Figs 1 and 2 present plots of $f$ against $t$ for leaching of $^{137}$Cs from the four grout samples, for Methods I and Method III.

5. Conclusion

Results are presented in Figs 1 and 2 which show the fraction of $^{137}$Cs leached from cement composites as a function of the square root of leaching period [3].

In the data for cement composite as a matrix, linearity between $f$ and $t$ (d) is not observed throughout the time tested; however, there are two different linearities before and after a leaching time of about 10 days. The slope of the linear relation for the early stage is larger than for the latter one. This change in the leaching rate may be associated with the fact that, as the leaching time elapsed, the diffusion rate would gradually slow down as the diffusion path becomes longer [3].

Method I cannot describe the whole leaching process, but it is very convenient to simulate leaching over a long period because of its simplicity. Despite the very complex numerical treatment required, the fit obtained using Method II is no better than that obtained using Method I. Although Method I is very similar to Method III, Method III provides the best approximation over the whole leaching period. In many cases, however, the leaching mechanisms are unknown and, therefore, it is convenient to use polynomial approximation.

The results presented in this paper give values that are similar to those reported in [3]. The solidification technique of spent mix bead (anion and cation) exchange resins is satisfied by cement immobilization.

Finally, the results presented in this paper will define the design of our future engineered trenches disposal system for radioactive waste.
FIG 2: Plot of fIII against t (d) for leaching of radionuclides from concrete (Method III)

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REFERENCES

Zero waste approach in treatment of NORM oil sludge wastes

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Abstract. Hazardous waste from the local processing and petroleum-based industries is increasing and various options of oil sludge waste treatment are needed. There is a need to explore new techniques for oil sludge treatment to a zero waste approach, i.e. a total solution of the problems. A brief description is presented where MINT has embarked to have various options available in the treatment of oil sludge that can be practiced in Malaysia. Environmental friendly chemicals are used to separate hydrocarbons from the oily sludge. Solid sediment materials which may consists of very low concentration contaminants such Thorium, Uranium and Radium, and possibly heavy metals, can be processed into allowable useful consumer products (glazed tile, building slabs, fuel material, pavement, etc.) in Malaysia. Other techniques (incineration) for decreasing the volume of oil sludge allow easy handling of the solid wastes. New techniques need to be explored by using effective microorganisms and plants to reduce toxicity and hydrocarbons in oil sludge. Hopefully, with guidance from MINT, these zero waste approach can be practiced in Malaysia in the years ahead.

1. Introduction
Various options of NORM oil sludge waste treatment in the local processing and petroleum-based industries need to be explored. Presently, oil sludge waste are not managed properly in Malaysia. There is an accumulation of wastes stored in drums in various crude oil and refinery terminals in Malaysia. Minimizing the volume of oil sludge wastes or recycle the products after treatment to ‘zero wastes approach’ is the best option to manage the wastes. The solid residues from the treated waste will be explored as consumers products such as, pavement/building slabs, foundation building materials, solid fuel, etc.

MINT has obtained permissions from the Department of Environmental (DOE) and Atomic Energy Licensing Board of Malaysia (AELB) to conduct research on the management of 2400 drums which contain oil sludge. The sludge contains naturally occurring radioactive materials (NORMs) and some heavy metals. There are different options to treat the oil sludge by chemical and separation treatment, chemical and physical processes (including incineration) and bioremediation processes.

Contaminants of interest in this study are the nonaqueous phase liquids (NAPLs) that are hydrophobic organic compounds that are slightly soluble or immiscible in water. Petroleum hydrocarbons and other organic compounds such as chlorocarbons are found at large numbers at government and industrial sites contaminating both soil and groundwater. The crude oil sludge can be considered as dense NAPLs which are normally obtained due to settling of deposits (crude oil sludge) which is oily, viscous tar-like material at the bottom of storage tanks at oil terminals. In general, NAPLs are frequent contaminants of soil and water. The crude oil sludge is considered as natural occurring radioactive material (NORM) which contains some naturally occurring radioactive materials such as thorium, uranium and radium radionuclides. NORM that is associated with oil production was discovered in 1981 in the North Sea (Smith 1987). In Malaysia, the awareness of NORM problem in oil and gas production industry started only in the late 1980’s. Eventhough the uranium and thorium concentration in oil sludge is very low, Smith (1987) stated that the more dominant radionuclides present are Ra-226 and Ra-228. The differences in this concentration may be due to the process chemistry whereby oil sludge is associated with produced water where radium salt has more preferential dissolution compared to uranium and thorium. This resulted in radium radionuclide being dominant in the oil sludge. Literature from several researchers (Weers et al. 1995, Testa et al. 1994, Kolb and Wojcik 1985, Miller et al. 1991) on average concentrations of Ra-226 and Ra-228 in oil sludge shows Malaysia oil sludge concentration (304 Bq/kg – 12,560 Bq/kg) is much lower than that of North Sea.
area (540,000 Bq/kg – 9,240,000 Bq/kg) and the United States of America (1,247,382 Bq/kg). Recently the topics on the management of technical enhanced naturally occurring material (TENORM) wastes have been given wide attention by the International Atomic Energy Agency and developed countries.

2. Zero waste approach

Treatment of oil sludge can be conducted in the laboratory on controlled pilot scale to separate or remove the dense non-aqueous phase liquid (NAPLs) in the oil sludge. The preferred method to treat the wastes will much depend on the chemical and physical characteristics of the oil sludge. Characteristics are the presence of radioactive material, heavy metals which are toxic and/or volatile such as lead and mercury metals and concentration of hydrocarbon. In the zero waste approach process, MINT and some local companies in Malaysia have conducted laboratory and small pilot scale treatment to remove the NAPLs or hydrocarbons from the solid residues/sediments using chemicals. Bioremediation method is another alternative where chemicals and additives were mixed to natural soil and oil sludges containing effective microorganisms and plants. With optimum conditions, this can fasten the breakdown of the organic materials in the oil sludge. By incineration the volume of the wastes is reduced.

In 2005, MINT will lead a research project to treat 500 ton oil sludge on a commercial scale by using a local oil sludge separation facility with chemicals supplied by another local company in Malaysia. MINT expects to obtain solid sediments residues of about 30 % of the 500 tons of oil sludge wastes. Research will be conducted on the solid sediments to convert them to consumers products to be recycled/reused rather than finally disposed. Recovered crude oil can be recycled as low grade oil.

3. Separation of crude oil from solid material by chemical sludge breakers

MINT and a local company, in 2003, have jointly conducted a laboratory and a small pilot scale research project on oil sludge minimization to separate hydrocarbons from oil sludge wastes. Chemical sludge breakers were mixed to oil sludge slurry to separate the hydrocarbons from solid materials. MINT has identified about 100 tons of oil sludge waste to be treated initially from a local Crude Oil Terminal, stored in rusted and old drums for the last 20 years. The graphical representation of the laboratory scale treatment of oil sludge and a flow chart for the small scale pilot process for the treatment is shown in Figure 1 and Figure 2 respectively.

From the small pilot scale oil sludge minimization process, achievements of the treatment project were verified by the samples that were validated and compared by various tests to the targeted Key Performance Indicators (KPIs) that were according to the standards set by the Department of Environment of Malaysia. This was to determine the treatment performed was achievable and the quality of the water effluent, oil and sediments collected were according to the local regulations.

The targeted KPIs were:

(a) 100 ppm ‘total organic content’ (TOC) in sediment
(b) 50 ppm ‘oil in water’
(c) 1 % ‘sediment in oil’.

From laboratory analysis, results on treated samples are:

(a) KPI value 100 ppm for ‘total organic content’ (TOC) in sediment; laboratory analysis on 10 samples: average 70 ppm - 90 ppm; below the KPI value of 100 ppm;
(b) KPI value 50 ppm for ‘oil in water’ in effluent; laboratory analysis on 10 samples: average 25 ppm – 35 ppm; below the KPI value of 50 ppm;
(c) KPI value 1% ‘sediment in oil’; laboratory analysis on 10 samples: average (< 1%) below the KPI value of < 1 %.

From the laboratory results, KPI values for TOC, oil in water and sediment in oil in samples, were below the KPI indicators, respectively. This indicates that the treatment process was successful in
separating the oil sludge into three parts such as oil, water and sediments and satisfy the allowable limits set by the Environmental Department. Effluent samples determined for specific radioactivity show that the effluent specific radioactivity was high between 0.8 Bq/l – 3.5 Bq/l for Ra-226. AELB regulations state that effluents with more than 1 Bq/l for Ra-226 cannot be disposed to the environment. The effluents need to be treated and carefully controlled before discharge to the environment. The hydrocarbon recovered can be recycled as low grade crude oil.

Sediments show specific radioactivity of radium concentration < 100 Bq/kg which are normal concentrations present in soil of Malaysia. However, one sediment sample shows a specific radioactivity of 922 ± 14.0 Bq/kg (Ra-226) and 782.8 ± 45.7 Bq/kg (Ra-228). These concentrations are high and need to be managed carefully according to the regulations set by the Atomic Energy Licensing Board of Malaysia. About 11 drums of sediments were recovered. In the zero waste approach, sediments collected from the treatment that are contaminated can be considered to be converted to consumers products. This is a different approach from previous waste management activities, stored temporarily and for eventual final disposal.

FIG. 1: Graphical representation of laboratory scale treatment of oil sludge
4. **High thermal treatment by incineration**

Incineration can be used to reduce high volumes of combustible oil sludge waste into smaller volume with a high reduction factor. Prior to the treatment, MINT has to conduct a detailed study to determine the behaviour of the NORMs and heavy metal when undergoing thermal treatment where possible to accumulate all the volatile NORMs and heavy metals in the fly ash. High thermal treatment on the hydrocarbon present in the oil sludge will reduce its volume but radioactive materials such as Thorium, Uranium and Radium still remain in the solid residues. The process produced bottom and fly ashes which contain NORMs and heavy metals. When burned, about 1 – 3 % by weight of the oil sludge is converted into fly ashes and about 70 % is in the form of bottom ash. It is more appropriate that the volatile NORM and heavy metals were accumulated in the fly ash. Fly ash formation with smaller volume will allow easier handling and management of the wastes.

5. **Bio-remediations of contaminated soils using effective micro-organisms and plant**

MINT has embarked on a bioremediation research project of soils contaminated with potentially toxic elements and oil sludge containing NORM using effective micro-organisms and plants. Indigenous soil micro-organisms included mycorrhizal fungi and local plant species. This process is to reduce toxicity and hydrocarbons in the oil sludge. The bioremediation methods are slowly but increasingly adopted, since they are regarded as “environment-friendly”. Use of plants or vegetation in remediation processes (phytoremediation) can enhance bioremediation processes as microbial transformations normally occur in the soil external to plant roots (Erickson *et al.*; 1993), including in soils contaminated with heavy metals (Pierzynski *et al*., 1993), organic contaminants e.g. polyaromatic
hydrocarbons (PAH), trinitro toluene explosive (TNT), oil spills, pesticides and sewage (Wiley, 1997) and radioactive isotopes of uranium, caesium and strontium from Chernobyl soils and uranium plant in Ohio (Schmiedeskamp, 1997; Wiley, 1997). Hazardous compounds and elements are transformed in plant tissue into less toxic forms (e.g. organics are transformed into CO2 and H2O after metabolism) or sequestered and concentrated so that they can be removed with the plant. Plants that have been used in remediation processes include alfalfa, poplar, bamboo, brassicas, compositaes, and tomato. Thus, interaction between plants and soil micro-organisms can be manipulated to their mutual benefit as well as to the well being of the environment.

6. Recycle and disposal options

Currently, MINT has two options to handle the solid residues from the treatment. Recycling the solid components into new products is the first priority in the zero waste approach. If there is no other alternative, final disposal into shallow land disposal or within a controlled area will be conducted.

6.1. Recycle

The solid residues from the treated waste can be explored as glazed tile, pavement/building slabs, solid fuel, etc. However, this new products must follow the radioactive controlled release limits for the use by consumers, set by the regulatory authority. Stringent regulations and experiments need to be conducted to justify the conversion of the materials into new products especially into household items.

6.2. Final disposal of treated oil sludge/solid residues from treatment

For disposal of treated solid residues from oil sludge, a radiological impact assessment (RIA) report must be produced in order to comply with the Atomic Energy Licensing Act 1984, Act 304 and the guidelines on radiological monitoring for oil and gas facilities associated with Technologically Enhanced Naturally Occurring Radioactive Materials (TENORM), LEM/TEK/30 SEM.2, September 1996. MINT has the experience in conducting the RIA study to determine the extent of the impact it causes on the population and the environment at the proposed site as a result of disposing treated oil sludge.

7. Conclusion

There is a need to explore new techniques for oil sludge treatment to a zero waste approach which is a total solution of the problems. Use of environmental friendly chemicals to separate hydrocarbons from the oil sludge allowe the solid materials which consist of the radioactive and heavy elements to be vitrified into useful product (glazed tile, building materials, slabs, solid fuel, etc.). New techniques need to be explored by using effective micro-organisms and plants to reduce toxicity and hydrocarbons in the oil sludge. Incineration decreases the volume of oil sludge and leads to easy handling of solid residues. Recycling the sludge oil products is the best option to manage the wastes. However, if there is no other alternative, final disposal can be conducted as long the regulatory requirements by AELB are fulfilled.

REFERENCES


Application of Safety Assessment Methodology to the Belarus “Military” Repositories of Well Type

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Abstract. After disintegration of the Soviet Union the repositories of radioactive waste (RAW) were abandoned in Belarus where the military divisions of the Russian Federation have been disposed before. For all that authentic information about constructional and operational characteristics of radioactive waste disposal sites, activity inventory, isotopic composition and distribution of radioactive materials into repositories was completely absent. Under preliminary inspection it has been established that these objects are the repositories of well type being intended for storage of disused radioactive sources of military-industrial origin. By now about 20 similar repositories have been found in Belarus. In 2001 the work program was developed to investigate some RAW repositories of the indicated type and to study their safety. In the paper the assessment of potential danger of one similar object, namely the “Kolosovo” RAW repository is given on the basis of consecutive application of safety assessment methodology (ISAM) having been developed by the IAEA experts. Analysis of field investigation data and calculational results connected with safety assessment of this object allowed to make the conclusion about necessity to do away with the object by restoring waste to another safe repository.

1. Introduction

After disbanding the military divisions of the Soviet Union in 1992-1994 the repositories of radioactive waste (RAW) of well type were abandoned in places of their former disposal in Belarus. For all that authentic information about constructional and operational characteristics of RAW disposal sites, activity inventory, isotopic composition and distribution of radioactive materials into repositories was completely away.

By now about 20 such objects have been found in Belarus [1]. As a result of preliminary inspection it has been established, that these repositories are the concrete cylindrical wells with buried disused radioactive sources there. The causes of anxiety regarding to these RAW wells were the following:

(a) High, not permissible levels of dose rate of gamma radiation on the upper surface of the source wells (300-2500 µR/h),
(b) Absence of technical detailed documentation of the well design, activity inventory, isotopic composition and isolation means of waste,
(c) Lack of any observation, control and registration at these dangerous objects,
(d) Necessity of analyzing potential danger of these repositories for present and future generations.

Retrospective analysis has shown that the objects considered do not have the official status and violate both the national and international requirements and regulations of RAW safe storage. In 2001 the work program was developed to investigate two RAW disposal sites of indicated type and to study their safety. In 2002-2003 this work was performed by the Joint Institute of Power and Nuclear Research-“Sosny” in co-operation with the Scientific-Research Geological Republican Facility “BelGEO” of NAS of Belarus.

Safety assessment of the “military” repositories was fulfilled on basis of the approaches which were developed under the harmonized research project (Improvement Safety Assessment Methodology for Near Surface Disposal Facilities (ISAM)) of IAEA and its successor ASAM project [2].

This paper is addressed the preliminary assessment of potential danger from the investigated “military” object, namely the “Kolosovo” RAW repository, following to main steps:
- Specification of the assessment context,
- Description of the repository system,
- Development of the scenarios to be assessed,
- Formulation and implementation of models and associated data, and
- Analysis of the results.

The general purpose is to test the applicability of the ISAM approaches and to use the results in future decision-making process.

2. Specification of the safety assessment context for the “Kolosovo” RAW repository

The “Kolosovo” radioactive waste repository was built in the 60-th of the last century in the Belarus territory of former disposal of military division of the Soviet Union. This object was destined for storing the disused radioactive sources of technogenic origin. After disbanding the military division as a result of the Soviet Union disintegration, the “Kolosovo” RAW repository found itself without a supervision and control by the state regulatory bodies. Preliminary inspection of the repository has shown, that this object was radiation dangerous because it did not correspond to both international and national requirements of RAW safe storage.

In line with this the general task of safety assessment was to determine the extent of potential danger of the repository considered and to make the decision either on its conservation, or strengthening engineer barriers for its reliable isolation from the environment, or its liquidating as radiation dangerous object and RAW restoring to another, more reliable repository. The specific purposes were:

- To assess the safety level of the object considered, using available information,
- To determine influence of uncertainty of the most important parameters on the safety assessment of RAW repository,
- To identify possible alternative models and scenarios which could be used in subsequent iterations of safety assessment,
- To help in decision-making on strategically important issues that concern to problems of its future existence, maintenance and control of RAW repositories of the type considered.

For all events an annual individual effective dose rate constraint of 0.1 mSv·y⁻¹ was applied as a radiological criterion.

3. Description of the repository system

The “Kolosovo” RAW disposal site is located in the Stolbcy district of the Minsk region. The distance to the nearest settlement is about 1400 m. The investigated place is on the right bank of the Neman river, in 7-10 km far from it. The operational water-supply boreholes used by population of the Stolbcy town and the nearest settlements are situated in 1250 m from the “Kolosovo” disposal site.

At the start of studying an authentic information about constructional and operational characteristics of the repository, activity inventory and isotopic composition of waste was completely absent. Besides nothing was known of the hydrogeological and geological conditions of the “Kolosovo” disposal site.

In order to get this knowledge, collaborators of JIPNR-Sosny NAS B developed the two years work programme on research of the “Kolosovo”RAW disposal site including the following tasks:

(1) Revealing the repository constructive features,
(2) Definition of activity inventory and isotopic composition of waste stored,
(3) Studying geomorphologic and hydrogeological conditions of the “Kolosovo” disposal site,
(4) Testing radioactive contamination of grounds and groundwater near the repository.

To decide on tasks 1 and 2 at first the control borehole was drilled at a distance of 0.8 m from the repository wall. When using the method of radiation carotage and sampling ground and groundwater under drilling, the following characteristics were determined:

- Depth of waste and repository,
- Rough isotopic composition of waste,
Radiochemical composition of ground and groundwater,
Presence of groundwater and its table.

When solving the last task the fund materials were utilized on disposal area, geological structures, hydrogeological and geological conditions of the “Kolosovo” disposal site. Next year the network of regimen boreholes has been installed for monitoring of radiation state of grounds and groundwater in the vicinity of the repository, and also for hydro-chemical sampling of near water-supple boreholes. On the basis of research performed, the component description of the RAW repository system was compiled.

3.1 The near field

The “Kolosovo” RAW disposal site is a near-surface repository of well type with simplified construction. The RAW base is in 6 m from land surface. The inspection has shown that the upper part of concrete well was not loaded by waste. As a result the total thickness of waste was 3.6 m.

Waste. Waste loaded in repository is kept in the containers. It is presented with disused radioactive sources of military-industrial origin. In repository the containers were untidy packed by layers and were filled up by sand. The total amount, packing and distribution of waste on the well depth were not known.

By means of spectra-metrical measurements inside and outside of the repository it was determined that the main radioisotopes in waste were $^{60}$Co and $^{137}$Cs. Sources with $^{90}$Sr were assumed to be contained in waste too. This fact is indirectly confirmed by $^{90}$Sr presence in the ground and groundwater samples extracted under the control borehole drilling. Using the results of gamma-spectra-metrical carotage research and calculations, the lower limit of $^{60}$Co specific activity was estimated. As a result of count again to moment of the last loading of waste (1994), $^{60}$Co specific activity was assessed as $3.46\times10^8$ Bq/kg. Specific activities of $^{137}$Cs and $^{90}$Sr in waste were estimated analogously, their values were equal about $3.46\times10^7$ Bq/kg. The upper limits of the specific activities were estimated to be distinguished from lower ones by two orders. The repository was supposed to be filled up with waste layers periodically from 1965 to 1994. The each waste layer was covered a sand one.

Engineered barriers. The “Kolosovo” RAW repository is the cylindrical concrete well with internal diameter of 1.0 m and wall thickness of 0.24 m. Upper part of well is risen above land surface at 2.3 m. It is banked up with sand ground covered by grass. The well is closed by a concrete lid with opening on which is mounted the removable metallic tumbler protecting repository from atmospheric precipitation. In connection with revealing radioactive contamination in the ground and groundwater samples near the repository it was supposed that the engineered barrier in the repository base was either absent or was destroyed to the moment of its investigation.

3.2 Geological and hydrogeological conditions

Area of the “Kolosovo” RAW disposal site is represented with naturally well drained flat fluvial-glacial plain. Thickness of fluvial-glacial depositions is 3.6 m. Depositions are presented with mid sand, slightly dump. The seasonal upper subsurface water can be accumulated in foot part of the depositions.

In the territory studied the morainic depositions are deposited under fluvial-glacial ones on the depth of 2-8 m and are presented in general by sandy soil and clay loams with gravel and pebbles. Thickness of morainic depositions is about 13.4 m. The first pressure aquifer is found on the depth of 11 m from the repository base. Hydraulic slope of the groundwater flow has a direction to a valley of the Neman river and is equal of 0.0023.

In researched area the unfavourable physical and geological processes are missing. The negative factor is that the well base is placed in an upper subsurface water zone. This furthers to washing away and leakage of radionuclides from repository to the environment.

3.3 The biosphere conditions

The “Kolosovo” RAW disposal site is situated in large wood tract represented by a pine, birch, oak, etc. Annual norm of atmospheric precipitation is 500-600 mm, annual average temperature is 7\degree C. The investigated area is on right bank of the Neman river, running in 7-10 km far from the repository. The
nearest water-supple borehole is located in 1250 m far from the object. Population utilizes surrounding territory for cultivating grain and vegetable cultures, growing cattle for meat and milk. The water from water-supple boreholes is used by population for drinking, watering land, drinking place of cattle and domestic needs of human.

4. Development and justification of scenarios

Based largely on expert judgement and use of the ISAM list of features, events and processes, three scenarios were considered: the Reference scenario and two Alternative ones. The **Reference scenario** envisaged the gradual release of radionuclides from repository as a result of their washing out from waste by percolating atmospheric precipitation. Migrating radionuclides in vadose zone, entering into aquifer, further transport by the groundwater flow to water-supple place and later to biosphere were also considered in the **Reference scenario**.

The **Alternative scenarios** describe some hypothetical events, which in the considered conditions are probable as a result of the future reorganizing of military division, in the territory of which the “Kolosovo” RAW disposal site is located. By this reason the object can find itself without control and guard, that can create prerequisite to destruction of external isolation, unsanctioned access to radioactive sources and application of object with purpose of terror or diversion. In line with this the following Alternative scenarios were considered:

1. Destruction or removing an external isolating overlapping; this can lead to direct access of atmospheric precipitation in repository and acceleration of migration processes,
2. Human intrusion.

5. Formulation and implementation of models and data

For the scenarios proposed were developed conceptual and mathematical models. Models in the **Reference scenario** and **Alternative No. 1** took into account the layer-by-layer waste loading into repository with period of 5 years under hydraulic stationary conditions, washing out radionuclides from waste by atmospheric precipitation and their migration with infiltrating moisture through engineer barrier and vadose zone up to aquifer, transport of radioactive contamination by the groundwater flow and later its entering to human as a result of water usage from water-supple borehole. The processes of radionuclide interactions in the systems RAW-water, ground-water, plant-water, human-environment and radioactive decay were taken into account too. In models of the **Alternative scenario No. 2** waste was supposed to be extracted from repository and scattered on surrounding territory, which can be hereinafter used in domestic farm for vegetable cultivation.

The structural scheme of mathematical model is represented by arbitrary set of vertical and horizontal boxes. The processes in boxes are described by the system of ordinary differential equations with lumped parameters, to be decided analytically. As initial information the results of experimental field and laboratory researches, data from fund and references were utilized [3, 4]. The test of model was performed on the basis of comparison of calculation results and measurements of the ground and groundwater contamination near repository considered and also in case of decontamination waste disposal sites of the Chernobyl origin [3].

6. Analysis of results

It was defined that the time of potential danger of the “Kolosovo” RAW repository is about 1000 years. The assessment of potential danger of this object showed that in the **Reference scenario** the most dangerous radionuclide is $^{90}$Sr. Conservative estimations testify, that the maximum specific activities of $^{90}$Sr in aquifer under repository can reach value $C_w = 8.4 \cdot 10^8$ Bq/m$^3$ in 200 years. In the place, where water-supple boreholes are arranged, $^{90}$Sr water contamination can reach a maximum value of $C_w = 1.12 \cdot 10^4$ Bq/m$^3$ in 320 years after closing repository and only in the distance of 2300 m this concentration will be decreased to the permissible level. This distance is the influence zone of the repository. Usage of the contaminated water by human for drinking and watering vegetables could lead to maximum individual dose rate exceeding 1 µSv/years during almost 100 years. The dose rate constraint of 0.1 µSv/years can be reached in 170 years after closing repository.
Calculations for the Alternative scenario No. 1, connected with destruction of external isolation overlap of repository through 50 years after its closing, showed, that this event could increase $^{90}$Sr concentration in groundwater and consequently the dose rate into the order.

According to the Alternative scenario No. 2, when intruding human in the repository system, extraction of waste and its scattering in surrounding territory could take a place. Usage of this territory for small farm could lead to annual individual dose rate from all exposure pathways to be equal about 200 Sv/year, which would be decreased to permissible level only in 500 years. In this case the main radionuclide forming dose would be $^{137}$Cs.

7. Uncertainty analysis

The research of uncertainty influence of the initial information on main calculational indicators of safety was carried out on the basis of a method of utmost estimation [3]. In this case the media parameters are known into some permissible range of their variations. As a rule the information about a probable distribution of media parameters into range is unknown. Then two utmost cases may be picked according to conservative (the most danger) and optimistic (the least danger) forecasts [3].

In the Reference scenario the most sensitive to the uncertainty analysis have appeared calculations for $^{90}$Sr. According to optimistic estimations $^{90}$Sr could reach the aquifer inappreciable concentrations (less than 0.1 Bq/m$^3$). But this would contradict the data of field investigations, which have fixed $^{90}$Sr presence in groundwater (more than 0.1 Bq/m$^3$) already in 10 years after closing repository. Thus it is possible to make a conclusion that in case considered the conservative assessment is more real than optimistic one.

The influence of uncertainty of model was studied on a specific example of its improvement by means of account of layer waste heterogeneity and dynamics of filling repository. First of all the registration of these factors had the influence on changing in time of radionuclide inventories and their distribution in repository, i.e. on near field. In this case the discrepancy of calculation results would be about 30-50%, and the greatest influence will appear in results for the Alternative scenario No. 2. When moving away from repository this influence will decrease and in aquifer the difference of radionuclide concentrations calculated by model considered models will not exceed about 3-5%.

The role of uncertainty under selection of scenario largely depends not only on a global evolution of events, but also from the human factor, namely from decision-making about future existence of repository.

8. Conclusion

The analysis of data of field investigations of ground and groundwater near the “Kolosovo” RAW disposal site has indicated to a start of processes of radionuclide migration from repository to environmental geosphere. Safety assessment of object, to be considered on the basis of approaches which were recommended by the IAEA experts (ISAM), also has given the conclusion about potential danger of this repository.

Parallel with available real danger of contamination of the environment near the “Kolosovo” RAW disposal site and similar objects, the problems of control of their state and guard arise in connection with forthcoming reorganization of military divisions. It can lead to unauthorized access to radioactive sources or diversion. In order to prevent these events, the performance of urgent measures is required, ultimate purpose of which will be liquidation of similar objects by means of restoring radioactive waste in more reliable repository.

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REFERENCES


Waste management policies and strategies in Brazil

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Abstract. The objective of this paper is to present, in a concise form, basic information about the radioactive waste management policies and strategies in Brazil, including the total inventory of radioactive waste in the country.

1. Introduction

The Brazilian Nuclear Programme comprehends the operation of several nuclear and radiation installations, as follows:

(a) two nuclear power plants in operation and one under construction (Angra 1 is a 657 MWe gross/626 MW net, 2-loop PWR and Angra 2, 1345 MWe gross/1275 MWe net, 4-loop PWR), all located in the Angra dos Reis City, Rio de Janeiro state. Angra 3 is a 1312 MWe gross/1229 MW net, 4-loop PWR and has had the construction temporarily interrupted since 1991. The two Brazilian reactors are operated by the engineering company Eletrobras Termonuclear S. A. (ELETRONUCLEAR);

(b) two uranium mine and milling operated by the Nuclear Brazilian Industry (INB). In Poços de Caldas city located in Minas Gerais state, a closed mine, has been operated from 1982 until 1991. All the economically recoverable Uranium has been extracted and currently no mining activity is under way. The Uranium treatment facility is still operational and has been used to process other source material from the second Brazilian mine, located in Caetité, Bahia state, with a capacity of treating 100t/y of U3O8. and the production of Lanthanum Chloride and Cerium Hydroxide. The new mining facility (Catetité) has been operational since 2000 with reserves of 100,000 t of U3O8, and a capacity of 400 t/year of yellow cake (U3O8), which can be expanded to 800 t/year;

(c) one fuel element complex named FEC located in Rio de Janeiro state also operated by INB including a reconversion plant and a fuel fabrication plant. The enrichment plant is expected to be in operation in the year 2005;

(d) four research reactors (the first and oldest in Latin America named IEA-R1 built in 1956 within the US Atoms for Peace program is located at the Nuclear and Energetic Research Institute-IPEN belonging to CNEN, on São Paulo University campus, in São Paulo city, with a maximum power of 5 MW and is used for physics experiments and radioisotopes production such as I-131, Sm-153 and Mo-99. The second named IPR-R1 is a 100 kW Triga reactor operating since 1960 at the Centro de Desenvolvimento de Tecnologia Nuclear (CDTN), also belonging to CNEN, on the campus of Federal University of Minas Gerais in Belo Horizonte and is mainly used for research work and the first fuel assembly replacement of the reactor is expected to occur only in the year 2010. The third Brazilian Nuclear Research reactor named ARGONAUTA (1965), also belonging to CNEN, is located at the Institute of Nuclear Engineering-IEN, on the campus of the Federal University of Rio de Janeiro, in Rio de Janeiro city, and can operate at a maximum power of 1kW/h. The low burn-up rate of the reactor (approximately 0.25 MW-Day) as in the case of IPR-R1, makes easier to storage the spent fuel. This reactor is used for training including sample irradiation. The last one, named IPEN MB-01, is also located at IPEN and is a the result of a national joint program developed by CNEN and the Navy and is basic a water tank type critical facility rated 100 W, and is mainly used for simulation of small LWR and research in reactor physics (its burn-up rate is below 0.1 MW-Day).
(e) a Pilot scale fuel cycle facilities, including a plant for the conversion of uranium to UF6 and another for uranium enrichment;

(f) 3248 medical, industrial and research facilities; and

(g) one industrial facility for processing of monazite sands.

2. Waste classification

According to the International Atomic Energy Agency (IAEA), radioactive wastes are classified in five categories. This system is also adopted in Brazil, as shown in Table 1 [1].

<table>
<thead>
<tr>
<th>Category</th>
<th>Characteristics</th>
<th>Disposal Option</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Exempt waste</td>
<td>Activity levels equal or below the exemption limit based on a maximum impact dose of 0.01 mSv/y for public</td>
<td>No radiological restriction</td>
</tr>
<tr>
<td>2. Low and intermediate level</td>
<td>Activity levels above exemption limit and heat generation equal or below 2 kW/m³.</td>
<td>Near surface repository or geological.</td>
</tr>
<tr>
<td>2.1. Short live</td>
<td>Long lived alpha emitters contents equal or below 4000 Bq/g and the average specific activity of all radionuclides in the package (immobilised) below 400 Bq/g.</td>
<td>Geological repository</td>
</tr>
<tr>
<td>2.2. Long live</td>
<td>Radionuclides alpha emitters concentration above limits cited before for short live</td>
<td>Geological repository</td>
</tr>
<tr>
<td>3. High level waste</td>
<td>Heat generation above 2kW/m³ and alpha emitters concentration above the limits allowed for low and intermediate level waste – short lived (2.1)</td>
<td>Geological repository</td>
</tr>
</tbody>
</table>

'Short lived' are those radionuclides with half-lives of approximately 30 years such as: Co-60, Sr-90, Cs-137, etc.

The major sources of radioactive waste in Brazil are at present:

(a) the Angra I and II Nuclear Power Plants;

(b) the Monazite Processing Industry which is being decommissioned;

(c) the 3,500 m³ of low level waste resulting from the decontamination work performed in Goiânia, following the 1987 accident that involved a 1375 Ci therapy source;

(d) the 3248 medical, industrial and research facilities that uses radionuclides.

The waste generated by the Uranium Mine and Milling Industrial Complex, although significant in volume, is kept at the site, in a dam specially built for this purpose.
3. Radioactive waste from nuclear power plants

The first Brazilian Nuclear Power Plant, Angra I, a two-loop pressurised water reactor of 626 MWe, Westinghouse design, in operation since 1981, has generated, until September 2004, a total of 2,012.4 m³ of solid/solidified wastes, with an accumulated activity of ≈304 TBq. Table 2 shows the quantities and percentages of LLW and ILW generated by Angra I-NPP and Table 3 from Angra II-NPP.

Since the Brazilian reprocessing programme has not been clearly defined, the Angra I spent fuel, stored at the on-site reactor basin (552 spent fuel elements) and from Angra II (151 spent fuel elements), poses, by far, the problem with the most difficult solution, in both technical and political senses.

![FIG. 1. Central Nuclear Almirante Alvaro Alberto – I](image)

Table 2: Nuclear Power Plant Radioactive Waste Inventory (Angra-I unit)-number of packages

<table>
<thead>
<tr>
<th></th>
<th>CA  a</th>
<th>EC  b</th>
<th>NCW c</th>
<th>SRP d</th>
<th>SRS e</th>
<th>COM f</th>
<th>IN  g</th>
<th>TOTAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>1982-2004</td>
<td>353</td>
<td>2,734</td>
<td>724</td>
<td>645</td>
<td>335</td>
<td>2,254</td>
<td>201</td>
<td>7,246</td>
</tr>
<tr>
<td>%</td>
<td>4.90</td>
<td>37.70</td>
<td>10.00</td>
<td>8.90</td>
<td>4.60</td>
<td>31.10</td>
<td>2.80</td>
<td>100.00</td>
</tr>
<tr>
<td>Volume (m³)</td>
<td>73.40</td>
<td>757.20</td>
<td>376.30</td>
<td>222.10</td>
<td>70.50</td>
<td>468.80</td>
<td>44.20</td>
<td>2,012.4</td>
</tr>
</tbody>
</table>

(a) ca-cartridges; (b) ec-evaporator concentrate; (c) ncw-non copressible waste; (d) srp-spent resin–primary; (e) srs-spent resin-secondary; (f) com-compressible; (g) in-inactive waste

Table 3: Nuclear power plant radioactive waste inventory (Angra-II unit)-number of packages

<table>
<thead>
<tr>
<th></th>
<th>EC b</th>
<th>COM f</th>
<th>TOTAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>2002-2004</td>
<td>76</td>
<td>48</td>
<td>124</td>
</tr>
<tr>
<td>%</td>
<td>39</td>
<td>61</td>
<td>100</td>
</tr>
<tr>
<td>Volume (m³)</td>
<td>9.6</td>
<td>15,2</td>
<td>24.8</td>
</tr>
</tbody>
</table>

4. Radioactive wastes from fuel cycle and monazite processing facilities

The uranium mining and milling industrial complex (CIPC), located at the Poços de Caldas plateau in the Brazilian state of Minas Gerais, has produced, from 1982 to 1991, 1170 tons of ammonium diuranate. The waste generated in this process is kept in a 29.2 hectares dam system, and with an actual volume capacity of 1 million cubic metres. It is estimated that 0.8 TBq (130 Ci) of U-238, 15 TBq (405 Ci) of Ra-226 and 4.2 TBq (112 Ci) of Ra-228 were disposed of in this site, to date. There
are, presently, about 600 metric ton of 'mesothorium' with an estimated Ra-228 activity of 1.85 TBq (50 Ci) disposed in a trench at CIPC and 0.2 TBq (6 Ci) also stored in a shed, at USAM in São Paulo (78 m³). The material with thorium hydroxide, separated from the rare earth elements during monazite processing, although not formally classified as waste, is also under storage in many installations in Brazil. The waste volume generated by the fuel elements assembly unit as well as by all the other pilot scale fuel cycle facilities is negligible when compared to the above-mentioned figures.

5. Radioactive wastes from medical, industrial and research applications

There are, at present, 3248 radioactive facilities in Brazil that use several radionuclides. Table 4 shows the number and percentage of these facilities, by field of application. It can be seen in this table that most of the radionuclides have industrial (≈30%) and medical applications (≈39%).

<table>
<thead>
<tr>
<th>Field</th>
<th>Number</th>
</tr>
</thead>
<tbody>
<tr>
<td>Medical</td>
<td>1261</td>
</tr>
<tr>
<td>Industrial</td>
<td>984</td>
</tr>
<tr>
<td>Research</td>
<td>694</td>
</tr>
<tr>
<td>Distribution of Radionuclides</td>
<td>61</td>
</tr>
<tr>
<td>Services</td>
<td>248</td>
</tr>
<tr>
<td>Total</td>
<td>3248</td>
</tr>
</tbody>
</table>

In 1989, the Brazilian Regulatory Body has suspended the authorisation given to several manufactures to use radioactive sources in lighting conductors. As a consequence, the estimated 75000 of these devices, installed all over the country, with an overall activity of the order of 3.7 TBq (100 Ci) of Am-241, will have to be received and stored by CNEN, in the next five years.

Most of the soluble radioactive wastes produced as a result of the use of short lived radioisotopes in medical institutions and research laboratories can be discharged into sanitary sewerage systems, after a given decay period, with concentrations and total activities not exceeding the limits specified in the Brazilian specific regulation. To be on the conservative side, it was assumed that the whole flask contents were not utilised. Moreover, in the determination of the recommended decay period, also taken into consideration was the time required for the specific activity of the contaminated empty flask (assumed to have 23 g and a remain with 2 % of the initial activity) to reach the value of 2 nCi/g (74 Bq/g), acceptable for dust bin disposal.

FIG. 2: Number of spent sources generated per state
Figure 2 shows the approximate percentages of the total spent sources (47,062), generated by several Brazilian States and, stored at CNEN’s Research Institutes, in percentage rounded (2,286 on the IEN-RJ institute, 4,756 CDTN-MG and 40,020 at IPEN-SP.)

6.   Radioactive wastes produced as a result of the Goiânia accident

The violation of a teletherapy source in Goiânia, Brazil, at the end of September 1987, with subsequent spread of most of its radioactive content, i.e. 1375 Ci of Cs-137, over a large urban area, brought about the need to estimate the quantities recovered during the decontamination work performed by CNEN. Approximately 3,500 m³ of wastes were generated, with an estimated overall activity lying between 47 TBq (1270 Ci) and 49.6 TBq (1340 Ci).

Taking into account the decay period necessary for the contents of all packages to reach a Cs-137 concentration level not greater than 87 Bq/g, it was possible to classify the drums and the metal boxes into 5 groups, as described in Table 5. Approximately 3,500 m³ of wastes were generated, with an estimated overall activity lying between 47,0 TBq (1270 Ci) and 49,6 TBq (1340 Ci) [2].

Table 5: Goiânia inventory

<table>
<thead>
<tr>
<th>Group</th>
<th>Time in years</th>
<th>Quantity metal boxes</th>
<th>Volume M³</th>
<th>Quantity drums</th>
<th>Volume M³</th>
<th>Total Activity TBq</th>
<th>Total Volume m³</th>
</tr>
</thead>
<tbody>
<tr>
<td>(1)</td>
<td>t=o</td>
<td>404</td>
<td>686.8</td>
<td>2,710</td>
<td>542</td>
<td>0.06</td>
<td>1,230.8</td>
</tr>
<tr>
<td>(2)</td>
<td>&lt;0 t &lt;=90</td>
<td>356</td>
<td>605.2</td>
<td>980</td>
<td>196</td>
<td>0.476</td>
<td>801.2</td>
</tr>
<tr>
<td>(3)</td>
<td>&lt;90 t &lt;=150</td>
<td>287</td>
<td>487.9</td>
<td>314</td>
<td>62.8</td>
<td>1.44</td>
<td>550.7</td>
</tr>
<tr>
<td>(4)</td>
<td>&lt;150 t &lt;=300</td>
<td>275</td>
<td>467.5</td>
<td>217</td>
<td>43.4</td>
<td>13.67</td>
<td>510.9</td>
</tr>
<tr>
<td>(5)</td>
<td>t&gt;300</td>
<td>25</td>
<td>42.5</td>
<td>2</td>
<td>.4</td>
<td>30.064</td>
<td>42.9</td>
</tr>
<tr>
<td>Total</td>
<td></td>
<td>1,347</td>
<td>2,289.9</td>
<td>4,223</td>
<td>844.6</td>
<td>45.71</td>
<td>3,134.5</td>
</tr>
</tbody>
</table>

Also the following packages were used in Goiânia:
- 1 metal package for the headstock with the remain source (4.4 TBq and with 3.8 m³);
- 10 ship containers (374 m³ with 0.4 TBq), and;
- 8 special concrete packages (11.4 m³ and 0.7 Bq).

The wastes were temporarily stored in open air concrete platforms occupying an area of about 8.5 x 10⁶ m² at a site near the village of Abadia de Goiás and 23 km away from the centre of Goiânia, a city with about 1 million inhabitants (Figure 3). It can be seen from Table 5 that approximately 33 % of the waste volume (Group 1) have specific activities not greater than 87 Bq/g. Furthermore, most of the recovered activity is distributed over only 16.5 % of the total volume, requiring a decay period greater than 150 years to reach acceptable concentration levels (Groups 4 and 5).

![FIG. 3. Provisional storage at Abadia de Goiás](image-url)
The remaining 40.8% of the waste volume (Groups 2 and 3), were placed into concrete containers to improve its conditioning as well as to provide an additional engineered barrier in the near surface repository to be constructed.

Although the specific activities of the wastes classified as Group 1 are inferior to the value established in the regulation for dust bin disposal of solid wastes by users of radioisotopes, this group will not be considered exempt from control. The Brazilian Regulatory Body understands that the above-mentioned exemption criterion was established for solid wastes generated by facilities that handle small quantities of radioactive materials. It is also understood that care should be taken to avoid the deliberate fractioning and dilution of wastes so as to achieve compliance with disposal regulations.

One near surface repository (CGP) was constructed in 1995 for all the radioactive waste below the 87 Bq/g limit (Group 1) after exemption from the Brazilian Environmental Licensing Body (IBAMA) and another one was constructed in 1996 for the radioactive wastes from groups 2-5 after licensing of IBAMA [3].

7. Waste management policy

According to Brazilian Legislation, the National Nuclear Energy Commission, CNEN, is the governmental body responsible for the receipt and final disposal of radioactive wastes in the whole country. The promulgation of regulations concerning waste management and disposal is also a responsibility of CNEN.

There are only, at present, in Brazil, two near surface disposal vault type facilities both constructed in Abadia de Goiás city (1995-1996), located in Goiânia state, for the safe disposal of the radioactive waste generated during the clean up operations in the Goiânia city due to the radiological accident of 1987 related to the violation of a teletherapy equipment containing Cs-137 used for cancer treatment.

On December 2001 a waste law was published in the country (N° 10.308) establishing the necessary set of requirements for the site selection of repositories, for the construction, licensing, surveillance, operation, liability, etc reinforcing the legal attributions of CNEN. The mesothorium (Ra-228 waste from monazite treatment) stored at CIPC, however, was placed in the local dam, since the studies conducted by CNEN have shown that the environmental impact due to this type of disposal is negligible. At present, the Angra I and II low level wastes are being stored at the nuclear power plant site. The high level wastes from the two nuclear Brazilian power plants are stored on the reactor pools waiting for a decision about the reprocessing possibility. The wastes from the radioactive facilities are stored in three CNEN's institutes of CNEN for waste treatment, as mentioned before.

For the low level wastes resulting from the operation of Angra-I and II nuclear power plants and from the use of radionuclides in medicine, industry and research, a technical discussion about the necessity of construction of a single national near surface vault repository (for all this LLW) or two different repositories are being analysed (one near surface vault for the LLW resulting from the nuclear power plants operations and a second one type borehole or vault near surface repository for the medical, industrial, etc LLW) are being studied. The location and design have not yet been established although some candidate’s sites have being studied in the country since 1980’s.

It is worth mentioning that political and psycho social aspects related to the subject of radioactive waste disposal (not in my backyard syndrome) contribute enormously to the difficulties faced by the Brazilian Government in the establishment of a national waste management policy. Two projects are being conducted by CNEN, in the field of safety assessment of final disposal facilities. The first project had the assistance of International Atomic Energy Agency (IAEA) with experts from many labs from the Department of Energy of United States (DOE) and the second is being conducted within the Federal University of Rio de Janeiro. The Federal University of Rio de Janeiro developed a national code for radionuclide migration and dose calculations for near surface repositories, and this work was finished at end of 1998.

The project with IAEA aims at establishing a national capability for assessing the safety of waste disposal facilities, and for this purpose, a multidisciplinary expert group was created- drawn from
CNEN's institutes and was trained in safety assessment methods, including the use of the relevant computer codes as well as laboratory and field measurements techniques.

In November 2002 the International Atomic Energy Agency (IAEA) launched a new co-ordinated research project in the field of safety assessment for near surface radioactive waste disposal facilities (ASAM – Application of Safety Assessment Methodologies for Near Surface Waste Disposal Facilities continuation of the ISAM project) with the participation of Brazilian experts.

The primary objective of the project is: to investigate the application of safety assessment methodologies used for post-closure safety assessment, in particular the methodology developed under the IAEA’s ISAM project (Figure 4), to a range of near surface disposal facilities; and to develop practical approaches to assist regulators, operators and other specialists in their review of such safety assessment [4].

REFERENCES


Radioactive waste management in Ghana

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Abstract. Radioactive materials have been used in Ghana for more than four decades. They are used in medicine, industries, and for research and teaching purposes. Due to the numerous social and economic benefits derived from the use of radioactive materials there has been a noticeable growth in the use of radioactive materials in Ghana. The radioactive waste produced must be managed. The National Radioactive Waste Management Centre was therefore established to carry out radioactive waste safety operations in Ghana. This paper highlights radioactive waste management strategies being adopted by the National Radioactive Waste Management Centre to manage radioactive waste anticipated and in existence.

1. Introduction
Work with radioisotopes in Ghana started in 1952 in the University College of the Gold Coast (now University of Ghana) with the application of radiostrontium on monkeys [1]. Following the success of these experiments, interest in radioisotopes application grew and with it, a general awareness in the country of the numerous economic benefits to be derived from application of radiation sources in diagnostic and therapeutic procedures in medicine, measurement and processing techniques in industry, irradiation techniques for food preservation and sterilization of medical products as well as for teaching and research. Statistics available indicate that over 30 institutions are using radioactive materials in Ghana.

The Government of Ghana through the Ghana Atomic Energy Commission (GAEC) established the Radiation Protection Institute (RPI) as the national regulatory authority in 1993 and was empowered by the Radiation Protection Instrument LI 1559[2]. With the increase in use of radiation sources in industries research and teaching and the establishment of a 30kW research reactor, 1850TBq gamma irradiator, radiotherapy and nuclear medicine units, the need for a national radioactive waste management facility become inevitable for the safe management of the wastes to be produced. Thus the National Radioactive Waste Management Centre (NRWMC) was established to carry out radioactive waste safety operations in Ghana. It was tasked to develop and establish a radioactive waste management infrastructure in Ghana. These include the establishment of an administrative structure, development of regulation and guidelines as well as the construction of a central waste processing and storage facility.

2. Legislative framework
To meet one of the fundamental objectives of radioactive waste management a National Radioactive Waste Management regulation has been drafted. A three-day seminar on Understanding and Implementations of the Regulation on Radioactive Waste Management in Ghana was organized for all stakeholders and decision makers. The initial draft regulation was reviewed by the RPI and discussed at the seminar. The final draft is receiving government attention for promulgation.

The regulation is based on separation of responsibilities between the waste management operating organisation (NRWMC), waste generator and the regulatory body (RPI). The primary responsibility for the safe management of the waste rests with the generator unless the responsibility has been transferred to the NRWMC. The waste generator is responsible for on-site collection, segregation, and temporary storage of the waste arising from his practice. The NRWMC is the designated national central facility with the requisite facility for the treatment, conditioning and storage of radioactive waste until a final disposal facility is established and becomes operational. The NRWMC will take control of the waste where the generator is incapable of handling the waste or the generator no longer
exists. The RPI is responsible for the enforcement of compliance with provisions of the waste management regulation and other relevant requirements by the waste generator and the NRWMC.

3. Waste management practices

Presently the radioactive wastes are generated mainly from research, medical and industrial use of radionuclides (table 1). All wastes that are not likely to decay to clearance levels within one year are transported to the NRWMC facility. NRWMC staff does collection of the radioactive waste from the waste generator. Transportation of the waste from the waste generation site is done in accordance with the International Atomic Energy Agency transport regulation. The NRWMC notifies the RPI of the transfer of spent or disused radiation source to the NRWMC facility. All wastes are accompanied by a document with detailed characteristics of the waste. Primary control of waste to check for acceptance criteria by NRWMC staff is carried out at the waste generation site. At the waste processing facility, an additional control and analysis is carried out, for treatment purposes. Decay storage is considered for short-lived low activity spent sources that will decay to clearance levels within one year from the time of its generation. Cement is used in conditioning operations in a 200L drum. The cemented waste is stored in the waste processing facility for at least 24 hours before transferring it to the storage facility.

Radium needles which were in use at the Radium Therapy Centre (RTC) of Komfo Anokye Teaching Hospital (KATH), Kumasi, for treatment of cervical cancer from 1992 – 1997 were conditioned with the help of the African Regional Co-operative Agreement (AFRA) Specialised Radium conditioning team from South Africa. A total of 190 mg (7.03 GBq) of radium was safely conditioned in one concrete package [3].

4. Facilities

A Central Waste Processing and Storage facility is been established for the treatment, conditioning and storage of radioactive waste generated in Ghana. The waste processing facility will have four operational areas and a laboratory. The four operational areas will be for receiving and segregation of waste, conditioning of non-compatible solid waste and sealed sources, controlled area for the conditioned waste drums, as the cemented waste will be stored in the waste processing facility for at least 24 hours before being transported to the storage facility. Although we do not expect to treat large volumes of liquid waste, a section of the facility will be made available to handle liquid waste. Research works to support technological processes in waste management will be carried out in the laboratory. A quality assurance programme will be developed and made operational, as it is an important requirement in a national waste management programme. The storage facility has the capacity to contain 100 200L drums. However the NRWMC has an operational interim storage facility, which consists of a room of 6.15m x 3m in dimension.

5. Disposal option

Statistics from the Registry Information System Database of sealed sources (table 2) shows that there will be considerable increase in disused and spent sealed radioactive sources in the country. There is therefore the need for the development of safe and cost effective final disposal solution taking into consideration the size and number of sources that need to be disposed. Ghana has therefore shown interest in the IAEA/AFRA Borehole disposal Of Spent Sources (BOSS) disposal concept been developed in South Africa. A group of scientists have been drawn from the National Nuclear Research Institute of G.A.E.C., to study the BOSS concept. The group will then in collaboration with scientists from Ghana Geological Survey Department and Water Research Institute of the Council for Scientific and Industrial Research to identify and characterise a site for the implementation of the BOSS concept when the concept has been validated by the International Atomic Energy Agency and found to be safe. They will undertake geological and geophysical studies to identify the structural and geological nature of selected areas. They will document the geochemistry and hydro-geology of the areas and will also provide sufficient data to enable an appropriate post closure safety assessment of the selected site to be done.

The BOSS disposal concept is been designed to provide a solution for the disposal of both short- and long life radionuclides such as $^{226}$Ra and $^{241}$Am. It consists of a standard 165mm diameter borehole
drilled to a depth of 100m with a 150 mm casing [4]. However wider and deeper (or shallower) boreholes will be drilled if required depending on site-specific conditions. The borehole will be sealed off with cement plug at the bottom to ensure that the disposal volume remains dry during the operational period. The disposal of the waste package will be limited to the bottom 50 m of the borehole and the rest backfilled with concrete to close the repository and prevent human intrusion.

Table 1: Inventory of Radioactive Waste in Interim Storage

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Half-life</th>
<th>Activity (mCi)</th>
<th>Type</th>
<th>Form</th>
<th>Quantity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sr-90</td>
<td>28y</td>
<td>2.0 – 10.0</td>
<td>Sealed</td>
<td>Solid</td>
<td>27</td>
</tr>
<tr>
<td>Cd-100</td>
<td>463d</td>
<td>3.0</td>
<td>Sealed</td>
<td>Solid</td>
<td>1</td>
</tr>
<tr>
<td>Ir-192</td>
<td>73.8d</td>
<td>5.0</td>
<td>Sealed</td>
<td>Solid</td>
<td>1</td>
</tr>
<tr>
<td>Fe-59</td>
<td>44.5d</td>
<td>100.0 -200.0</td>
<td>Sealed</td>
<td>Solid</td>
<td>2</td>
</tr>
<tr>
<td>I-129</td>
<td>1.6E7y</td>
<td>125.0</td>
<td>Sealed</td>
<td>Solid</td>
<td>1</td>
</tr>
<tr>
<td>Co-57</td>
<td>271.8d</td>
<td>1.0</td>
<td>Sealed</td>
<td>Solid</td>
<td>1</td>
</tr>
<tr>
<td>Co-60</td>
<td>1925.5d</td>
<td>1.0</td>
<td>Sealed</td>
<td>Solid</td>
<td>1</td>
</tr>
<tr>
<td>Sr-89</td>
<td>50.5d</td>
<td>129.0</td>
<td>Sealed</td>
<td>Solid</td>
<td>1</td>
</tr>
<tr>
<td>Ti-204</td>
<td>3.7y</td>
<td>1.0</td>
<td>Sealed</td>
<td>Solid</td>
<td>2</td>
</tr>
<tr>
<td>Cs-137</td>
<td>30.0y</td>
<td>1.0 – 350.0</td>
<td>Sealed</td>
<td>Solid</td>
<td>30</td>
</tr>
<tr>
<td>Cf252</td>
<td>2.64y</td>
<td>20.0</td>
<td>Sealed</td>
<td>Solid</td>
<td>6</td>
</tr>
<tr>
<td>P-32</td>
<td>14.3d</td>
<td>10.0</td>
<td>Unsealed</td>
<td>Liquid</td>
<td>1</td>
</tr>
<tr>
<td>Tc-99m generators</td>
<td>0.3d</td>
<td>135</td>
<td>Unsealed</td>
<td>Liquid</td>
<td>77 (decommissioned)</td>
</tr>
<tr>
<td>In-113m generators</td>
<td>1.66h</td>
<td>50.0</td>
<td>Unsealed</td>
<td>Liquid</td>
<td>12</td>
</tr>
<tr>
<td>Ra-226</td>
<td>1600y</td>
<td>10mg</td>
<td>Sealed</td>
<td>Solid</td>
<td>19 (conditioned)</td>
</tr>
<tr>
<td>Am-241</td>
<td>432.2y</td>
<td>1.0 – 10.0</td>
<td>Sealed</td>
<td>Solid</td>
<td>125</td>
</tr>
<tr>
<td>C-14</td>
<td>5730y</td>
<td>1.0</td>
<td>Sealed</td>
<td>Gaseous</td>
<td>7</td>
</tr>
</tbody>
</table>

Table 2: Inventory of licensed radiation sources in use in Ghana

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Quantity</th>
<th>Total Activity (Bq)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Am-241</td>
<td>11</td>
<td>5.50E+10</td>
</tr>
<tr>
<td>Am241-Be</td>
<td>18</td>
<td>3.37E+10</td>
</tr>
<tr>
<td>C-14</td>
<td>1</td>
<td>3.70E+07</td>
</tr>
<tr>
<td>Cd-109</td>
<td>3</td>
<td>3.33E+08</td>
</tr>
<tr>
<td>Cm-244</td>
<td>3</td>
<td>6.40E+09</td>
</tr>
<tr>
<td>Co-60</td>
<td>6</td>
<td>2.27E+15</td>
</tr>
<tr>
<td>Cs-134</td>
<td>3</td>
<td>1.29E+09</td>
</tr>
<tr>
<td>Cs-137</td>
<td>53</td>
<td>2.39E+11</td>
</tr>
<tr>
<td>Fe-55</td>
<td>1</td>
<td>7.40E+08</td>
</tr>
<tr>
<td>H-3</td>
<td>4</td>
<td>4.44E+09</td>
</tr>
<tr>
<td>Ir-192</td>
<td>6</td>
<td>1.48E+13</td>
</tr>
<tr>
<td>Ni-63</td>
<td>1</td>
<td>3.70E+08</td>
</tr>
<tr>
<td>Pu-238</td>
<td>1</td>
<td>1.11E+09</td>
</tr>
<tr>
<td>Ra-226</td>
<td>1</td>
<td>1.48E+09</td>
</tr>
<tr>
<td>Sr-90</td>
<td>21</td>
<td>1.41E+10</td>
</tr>
<tr>
<td>Y-90</td>
<td>7</td>
<td>2.59E+08</td>
</tr>
</tbody>
</table>

6. Conclusion

In this paper the national radioactive waste management infrastructure has been highlighted. Promulgation of the national radioactive waste management regulation and completion of the central waste processing and storage facility will provide a solid foundation for the safe management of radioactive waste in Ghana. A financing system must be created to help sustain the waste management activities, especially research in disposal options.

REFERENCES


Planning closure safety assessment for the Egyptian near surface disposal facility

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Abstract. The development of a repository is a significant national effort requiring several decades to complete, as well as a substantial amount of skilled human, economical and technical resources. Planning and implementation of different disposal activities proceed in a stepwise manner, guided by decision points. Granting license is considered one of the major decision points that require the presentation of a safety case to evaluate the acceptability of the repository practice. The Egyptian regulation requiring the conduction of safety assessment studies as an essential requirement in license application for construction, operation, and closure of a near surface disposal. Closure of a disposal facility is the last major operational step in completing the disposal system. It requires the consideration of a combination of scientific, engineering, regulatory, and socio-economic factors that are integrated and optimized to select cost-effective alternatives acceptable to all interested parties. In this work, a general framework will be established to conduct a safety assessment methodology for closure of near surface disposal.

1. Introduction

Waste disposal facilities are designed with the primary aim of containing and isolating radioactive wastes. Since complete containment and isolation cannot, in practice, be guaranteed for the whole period that the waste presents a potential hazard, a second aim is to ensure that any eventual releases do not present an unacceptable risk [1].

Stepwise planning and implementation of different disposal stages are aspects of project management. At the beginning, many decision points are considered, which include the definition of the types and amount of waste to be disposed of, and the choice of host rock, engineering design, and sites to be investigated. Once a site is identified and an initial engineering design is defined, a more detailed planning the scope of above and below-ground investigations is required. Then a more mature plane will be focused on obtaining any necessary legal or regulatory approvals for construction, operation and eventually, closure. A detailed safety assessment and the presentation of a safety case in the form of a structured set of documents are typically required for granting licenses. A license to construct, operate, and close the disposal facility, will be granted only on the condition that the overall disposal system produce an acceptable safety case. Figure 1 shows the lifecycle of the disposal system including site selection, licensing, construction, operation, closure, and post closure [2].

Disposal closure includes both administrative and technical actions whose purpose is to complement the design of the disposal system. It is defined as a set of systematic actions that is conducted after the receipt of waste ceases and waste emplacement operations have been completed with the intention of providing a final configuration for the disposal system [3]. Repository closure is complete when the regulatory body, confirms that the closure activities have been performed in an acceptable manner, that the appropriate documentation is available and that provision has been made for post-closure controls.

2. Closure phase activities

In general, disposal facilities are required to have closure systems that provide control on ingress of water and to protect the workers, and members of the public against hazards posed by the waste. The first step in the closure phase is to submit a preliminary closure plan to regulatory body, it should includes detailed description of the closure method, updated safety assessment for the facility, monitoring and surveillance plan, description of the record keeping and record preservation system, and long-term controls that will be implemented during the post-closure phase.
Closure plans should reflect any “as built” modifications to the original facility or operational practices that would affect the results of safety assessments. The primary objective of the closure plan is to satisfy regulatory requirements and address public concerns to achieve safe isolation of the waste in a cost-effective manner. The closure system is expected to function with minimal maintenance, without losing integrity, by promoting drainage to minimize erosion, infiltration, and accommodate settling and subsidence.

3. Planning for up-dating safety assessment

The long-term safety of the waste disposal system is generally evaluated by conducting safety assessment studies [4]. Closure systems are subsystems of the overall waste disposal system, and thus are evaluated in concert with the overall safety assessment. The safety assessment should ensure that the safety of the overall disposal system is preserved during the operational period, and the integrity of the waste isolation did not change. It should also include how the radioactivity will be monitored, and the updated prediction of the impact of repository on man and the environment.

The general framework for conducting an updated safety assessment is illustrated in Fig.2.
The up-dated safety assessment should identify a general set of performance criteria for the closure system (e.g., limits for infiltration rates, service lifetime required of cover, requirements to mitigate intrusion risk). This should include the following elements:

- The assessment context of the safety assessment should be related to the evaluation of the effects of the actual source term and various components of the closure system on the safety of the overall disposal system and if the system will preserve its safety adequacy or not.
- The detailed description of the overall disposal system including the actual source term, physical form and chemical composition of the waste, and the closure systems;
- Identification of closure features, events and processes that could influence release and transport rates from the waste to the geosphere and biosphere to humans, and the definition of scenarios of critical importance;
- Evaluation of the effects of various components of the closure system and scenarios against the base case PA for the disposal facility;
• Consideration of the uncertainties in the results and identification of the parameters and assumptions that are of most importance (“uncertainty and sensitivity analysis”); and comparison of results with the appropriate performance standards and regulatory criteria.

4. Conclusion

The closure safety assessment is required to ensure that the safety of the disposal system is preserved, and to take into account changes in the waste inventory occurred during the operational phase. It should also indicate how the system will be monitored, and up-date the prediction of the impact of the repository on man and the environment.

Acknowledgement. Gratitude is expressed to Prof. Dr M. R. El-Sourougy for his leadership to the safety assessment team at the Hot Laboratories Centre and for his thoughtful comments.

REFERENCES

Waste management policy and disposal concept design for LLW in Turkey

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Abstract. Interim storage is the current situation of conditioned LLW in Turkey. Long-term disposal of LLW will be the next stage in waste management policy of Turkey. Waste management policy and strategies are based on the protection of human health and the environment. For this purpose, TAEA has started to develop near surface radioactive waste design concepts to assess and demonstrate the safety of the waste disposal facility. In this paper, main disposal options, disposal programme, description of the system, design parameters and preliminary basic data are presented. Several design studies of near surface facilities in several sites have been taken into account for safety assessment. In addition of the advantages of vault disposal, advantages of borehole disposal has been taken into consideration especially for spent sealed sources in Turkey.

1. Introduction

The waste management policy of TAEA, based on waste management principles, covers: protection of human health; protection of the environment, also beyond national borders; health of future generations, i.e. not to impose undue burdens on future generations; establishing national legal framework; minimum waste generation; interdependencies among all steps; maintaining safety of waste management facilities. Currently, predisposal waste management activities have been done at ÇWPSF in Istanbul. Liquid LLW have been immobilized into drums. Solid waste has been conditioned into drums and all of the spent sealed sources have been stored in original shelters or immobilized into drums. Radium sources and long-lived radionuclides have been stored as retrievable in drums. We are in site selection (clay and tuff) and design studies (vault type) stages for near surface disposal facilities. We will pass several distinct phases including site selection, facility construction, disposal operations and final site closure. For this reason, we are planning to make safety assessment studies on our candidate sites and our candidate designs to support final disposal decisions.

2. Waste management policy

Disposal strategies. According to the waste management policy of TAEA, disposal strategies will be forced. These strategies aim to provide maximum radiation protection of members of the public following the disposal of radioactive waste. For this purpose, reasonable measures will be taken to satisfy the constraint for natural processes and to reduce the probability of human intrusion. For safety assessment; two categories of exposure situations are considered; natural processes and human intrusion. Generic reference levels are in natural processes < 0.3 mSv/year and in human intrusion <10 mSv/year. During the disposal concept design and the site selection process, assessment of predicted performance data and experimental performance data has been planned for safety assessment.

Waste inventory. The total amount of the waste inventory is not too big, the majority of waste being spent sources. The main radionuclides and their migration characteristics have been investigated, users are known and projections about the future capacities can be estimated. The main radionuclides taken into consideration, including future waste generation, are: 3H, 10Be, 14C, 22Na, 41Ca, 54Mn, 55Fe, 59Ni, 63Ni, 60Co, 65Zn, 90Sr, 93Zr, 94Nb, 99Tc, 106Ru, 110mAg, 121mSn, 126Sn, 125Sb, 129I, 134Cs, 137Cs, 144Ce, 147Pm, 151Sm, 152Eu, 154Eu, 204Tl, 210Pb, 226Ra, 228Ra, 227Ac, 232Th, 234U, 238U, 237Np, 238Pu, 239Pu, 240Pu, 241Pu, 241Am.
3. Disposal concept design

The safety assessment will provide a decision-making tool. Site specific data will be used and all potentially significant factors will be considered. Designs of the facility will be based on projected low level radioactive waste arising from medical and research institutes and industrial applications over a 50 year period. Candidate sites are under investigation. All the wastes have been stored in an interim storage facility. Although a final decision has not yet been taken, main disposal options for conditioned LLW are shown in Figure 1.

![FIG. 1: Main disposal options](image.png)

Both disposal methods have been taken into consideration. Near surface disposal has especially low cost per waste package and the total capacity of the disposal facility is convenient for long-term disposal.

![FIG. 2: Disposal programme](image.png)
On the other hand, technologies used during borehole construction help to make engineered barriers with desired parameters, which meet geological conditions of a given site. Such barriers will provide reliable isolation of radioactive wastes. Monitoring systems will provide the possibility to control conditions during waste storage both inside the repository and outside, and allow remedial maintenance in case of an emergency. Hoisting mechanisms and charge-discharge systems allow removing waste packages at the end of the storage period or in the case of a need to retrieve the containers.

**System description.** Geography, meteorology, climatology, geology and hydrogeology of several candidate sites have been investigated. One of these sites has, 20-80 meters from the surface, pure clay or marl type clay based formations, and the underground water level is between 40-80 from the surface. Alternative sites are also under investigation. The investigations are based on the tuff and ultra basic formations especially for the long term disposal. Radionuclide migration experiments have been investigated in situ and laboratory tests within the national project concept.

**Specifications of potential sites:**

- Saturated Hydraulic conductivities \(2, 0 – 5, 5 \times 10^{-3} \text{ m/d}\);
- Porosities (0, 3-0, 5);
- Unsaturated zone (extends 60-100 meters below the surface);
- Long term average rainfall (70-120 mm per annum);
- Average natural background radiation level (2, 00-2, 5 mSv/year).

**Basic Specifications.** Conditioned radioactive waste, grouting is planned; drums and containers, 1 m cap, upper cap slope is 5 %; drainage system is active during operation; inspection wells. Design parameters: in current conditions, a small near surface facility will be sufficient for the next 30-50 years. General design studies focused on vaults. Each vault is 2.5 m x 6.5 m, and 5 concrete vaults in each rows, each disposal unit has total 25 vaults. Design details (concrete cap, exact volume, package forms, drainage and other construction details) can be changed according to the safety assessment.

**Operational phase.** Acceptance criteria have been started from the waste package properties at the waste immobilization stage. Concrete vaults will be used for disposal. Eighty drums (4x10x2 = 80 drums) per one concrete vault. Closure and institutional control; using of bentonite added concrete covers and diatomite barriers. Waste acceptance requirements have been specified in order to ensure that the disposed waste packages compromise the safe confinement of the waste by diatomite barriers. According to waste acceptance criteria, the institutional control period is determined as 100 years. Waste packages, regulatory framework and regulations are based on international principles (e.g. ICRP 60, 77 and 81). Following the closure of the disposal facility, it will be for 100 years under active institutional control (security fences, environmental monitoring, maintenance etc.) for maintaining the integrity of the disposal system. Passive institutional control (records, map marks etc.) will last for a longer period (150-300 years).
Radioactive wastes will be managed in such a way that estimated impacts on future generations will not be greater than today. Therefore, safety assessments will be developed according to the lifetime of the disposal facility. In addition, regulatory control will be based on the determination of the safety of the facility on estimates of dose but to consider other criteria, including natural events (e.g. earthquake for Turkey). The assessed radiological impact of disposal to members of the public will be consistent with dose constraints, which are to be some fraction of 1 mSv/y for normal exposures. Consistent with ICRP 81 [2], the protection of the public from the long-term impacts of the disposal facility do not exceed 0.3 mSv/y for normal exposures. It is assumed that human intrusion exposures lower than 10 mSv/y should not warrant further consideration. The post closure assessment will be based on current human behaviour, habits and actions at the potential sites. Our safety case will only consider inadvertent human intrusion.

4. Conclusion

All LLW has so far been stored in interim storage. TAEA has started to investigate near surface disposal methods for long-term disposal of LLW. According to the waste management policy and strategies of TAEA, preliminary objective in this field is ensuring safety assessment and the structuring of safety cases, supported by convenient regulatory developments. Various disposal options have been taken for further investigation. These include not only several near surface disposal design concepts but also borehole and LDW (large diameter wells) disposal as well.

REFERENCES


Basic elements of confidence in each stage of a new near surface repository safety assessment

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bBucharest University, Faculty of Geology and Geophysics, Bucharest, Romania

Abstract. In June 2004, at Budapest, the IAEA hosted the first workshop dedicated to safety case development for near surface waste disposal facilities: “Regional Workshop on Structure and Content of Safety Cases and Development of Confidence in the Safety of Near Surface Waste Disposal Facilities”. This paper deals with the safety case key components for a new near surface repository, particularly for the early stage of repository development, following the basic information presented in the frame of this workshop. In this stage of the repository life cycle the main purpose of the safety case (preliminary safety case) is to demonstrate that the plant is capable to be constructed and operated safely. The paper will analyse the basic elements of confidence in safety assessment: confidence in the safety assessment methodology, confidence in the safety assessment approach, and confidence in each stage of the safety assessment. We shall discuss the confidence building process through all stages of the safety assessment of a new near surface disposal facility. Each of these steps will be examined using the steps recommended by the ISAM methodology: assessment context, description of the system, development and justification of scenarios, formulation and implementation of models, analysis of the results, etc. To enhance the confidence in the safety assessment results, we considered important to demonstrate a good understanding of the phenomena, mathematical models and numerical methods involved in the safety assessment. Also, the confidence in the safety assessment results can be enhanced by demonstrating compliance with the regulatory requirements.

1. Introduction

According to the Joint Convention, the Contracting Parties are obliged to assess the safety of waste management facilities prior to their construction and operation and to review the safety of existing facilities. There are continuing international initiatives to develop, improve and harmonize approaches to assessing and demonstrating the safety of waste disposal facilities, such as the Agency’s ISAM (Improving Safety Assessment Methodology) and ASAM (Application of Safety Assessment Methodology) programmes.

The safety assessment alone is not sufficient to demonstrate safety to the satisfaction of all stakeholders involved in the development of the disposal facility. Confidence in the long-term safety is promoted and communicated through the safety case, which is more than results of safety assessment calculation it includes both qualitative and quantitative arguments. It is preferable that a separate safety case is produced for each stage of repository development that should demonstrate the safety of that stage before it commences and should be forward looking to subsequent stages [1].

The aim of this paper is to emphasise the fundamental aspects related to confidence building in the context of a safety assessment in the early stage of repository development. In the first part of the paper the overall confidence in the safety assessment methodology and approach are pointed out. Finally, the confidence building process through all stages of the safety assessment is presented. Each of these steps will be examined using, the steps involved in the ISAM methodology.

2. Confidence in the safety assessment

The purpose of preliminary safety case for a new near surface repository is to demonstrate that the plant in principle is capable of being constructed and managed safely throughout all life cycle from conception up to decommissioning.
At early stages of development/assessment there will be relatively little site-specific information available. It is essential to consider whether the existing amount of information is sufficient to allow the demonstration of conservatism and to support a decision for this stage of the safety assessment [2].

2.1. Confidence in the safety assessment methodology

The safety assessment methodology should allow the regulators and other technical reviewers to easily reproduce the assessment results, follow the logic and understand the assumptions used in the assessment [3]. Confidence in the safety assessment is increased if the methodology is compatible with international practice. Because the early stages of assessment are characterized by a general lack of data, stylised presentations are useful. These include, for example: reference biospheres, human action scenarios and reference critical groups.

2.2. Confidence in the safety assessment approach

At early stages of assessment the simplified calculations are needed for overcoming uncertainties arising from incomplete description of the system or incomplete data [4]. The conceptual and numerical models that are considered to be most likely in light of the available site-specific information were preferred. We used the simplified calculations to simulate release and transport of radionuclides from the near-field. For example, we considered one disposal cell having the dimensions of 15.50 x 29.00 x 8.00 m and the total C-14 inventory uniformly distributed into this cell.

The treatment of uncertainties is carried out by analyzing of a suite of variants with alternative parameter combinations or conceptual models. In context of little site-specific information for conceptual and mathematical model, probabilistic methods were based on site specific and generic probability density functions.

2.3. Confidence in each step of the safety assessment

The confidence building process through all steps involved in the ISAM methodology are examined for the preliminary safety case. At this early stage of the safety assessment it is necessary a complete understanding of the key components of the assessment context [4].

For this stage of the repository development, the regulatory framework is based on broadly accepted international principles and the confidence can be built by demonstrating an understanding of regulatory requirements. In this study a dose limit of 1 mSv/y is used for demonstrating compliance. Regarding the assessment philosophy, a combination of probabilistic and deterministic approaches was applied in the most analyses.

For system description confidence building it is necessary to ensure that the collected data are pertinent to the assessment context and to document the assumptions made and the associated uncertainties. The present study is based on investigation program at Saligny site. The concept of disposal system is similar to that in other countries being based on a multibarrier system, which combines the natural geological environment with engineered barriers. The disposal system concept capitalizes a particular area, the closeness to the Cernavoda Nuclear Power Plant, and a deep unsaturated zone. The geology of the unsaturated zone consists of in a layered sequence with the following identified horizons: silty loess (~ 30 m), red clay (~ 8 m) and pre-quaternary clay with sand and limestone lens (~ 12 m). The unsaturated zone overlays a confined aquifer consisting of limestone of barremian age with a variable thickens, between 20-40 m. It was assumed that the geosphere interface with the biosphere is a well located at the down gradient edge of disposal cell.

A systematic approach was adopted for defining scenarios, and all relevant FEPs that could influence the performance of the disposal system were included. To estimate the long-term repository performance under abnormal conditions arising from human activities, in this paper we have analized the resident farmer scenario which is credible to occur in the future. This scenario in which a family constructs a home on the contaminated site and raises an appreciable fraction of its food on site, is considered to be a credible bounding scenario because on-site residents receive a dose that is at least as large as the dose to off-site residents and is generally larger.

In order to build the confidence in conceptual models it is essential to demonstrate that the conceptual models and associated data are consistent with the assessment context, the disposal system,
scenarios to be investigated, and the software tools in which the mathematical models are encoded. For preliminary safety case the data needed for detailed modelling are not available and the use of simple models represents a normal alternative. The models employed provide a clear support for safety case by including a balance of simplicity, realism and conservatism. The model used is based on concentration factor method where the relationship between radionuclide concentrations in soil and the dose is expressed as a sum of the products of pathways factors.

The treatment of parameter uncertainty was quantified using the deterministic and probabilistic methods. Inside of deterministic approach, the behaviour of the disposal system under various conditions was evaluated by analysing a suite of variants where alternative parameter combinations were used. In this manner, the treatment of uncertainty is carried out by conducting a set of deterministic analyses and by choosing one of them as basis for the decision. Following the deterministic approach we used the sensitivity analysis to rank the most influential parameters in the assessment. In this context, we calculated for which input parameter the dose shows a significant variation for a given reasonable variation of the value of that input parameter. Probabilistic methods consist of a series of deterministic analyses and the relationships between the deterministic results are expressed using probability theory. In the framework of this approach the basis for the decision is the probability of exceeding a performance objective. In this study the input data needed for the sample generation include the initial seed value for the random number generator, the number of observations, the number of repetitions, the sampling technique, the method of grouping the samples generated for the different parameters, the type of statistical distribution for each input parameter, the parameters defining each of the distributions, and any correlation between input parameters [5].

3. Results

In this chapter we only presented the results of sensitivity and uncertainty analyses performed with RESRAD V6 [6] and related to the input parameters. All the calculations are based on a conservative model for near field, where the critical radionuclide C-14 is considered uniform distributed in the disposal cell at the beginning of the simulations and having a very low distribution coefficient. Sensitivity analysis showed that the dimensionless sensitivity dose coefficients with respect to drinking water intake (0.961), distribution coefficients in contaminated zone (0.985) and unsaturated zone (0.554) are much larger than, for example, sensitivity coefficient of density of contaminant zone (0.037). The deterministic and probabilistic approaches were employed for quantitative uncertainty analysis in the results of safety assessment.

The treatment of uncertainty in the deterministic approach is based on the reference case considered to be a conservative interpretation of the system performance. The behaviour of the system under the effects of uncertainties in the input parameters was evaluated by analysing a suite of variants where alternative parameter combinations were used. These variants might be caused by uncertainties in the input parameters but the likelihood of different variants cannot be evaluated in quantitative terms. The results of deterministic approach are shown in Table I.

<table>
<thead>
<tr>
<th>Variant</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dose (mSv/y)</td>
<td>8.85E-01</td>
<td>9.37E-02</td>
<td>3.85E-02</td>
<td>1.01E-02</td>
<td>6.04E-01</td>
<td>1.14</td>
<td>1.22E-01</td>
</tr>
</tbody>
</table>

The probabilistic analyses assign probability density functions to individual parameter values, which allows for sampling of inputs and generation of a distribution of consequences. This sampling implies that many calculations are necessary to achieve satisfactory output statistics. The probabilistic analysis was performed on a total of seven input variables: indoor and outdoor time fractions, drinking water intake and four distribution coefficients. Three rank correlation coefficients were specified between the indoor and outdoor time fractions and drinking water intake [6]. The results of probabilistic analysis are shown in Table II.
Table II: Results of the probabilistic analysis

<table>
<thead>
<tr>
<th>Measure</th>
<th>Dose (mSv/y)</th>
<th>+/-</th>
</tr>
</thead>
<tbody>
<tr>
<td>Min</td>
<td>2.96E-02</td>
<td></td>
</tr>
<tr>
<td>Max</td>
<td>3.29E+00</td>
<td></td>
</tr>
<tr>
<td>Mean</td>
<td>3.36E-01</td>
<td>1.56E-03</td>
</tr>
<tr>
<td>Standard Deviation</td>
<td>2.69E-01</td>
<td>9.31E-03</td>
</tr>
<tr>
<td>50-th Percentile</td>
<td>2.55E-01</td>
<td>6.95E-03</td>
</tr>
<tr>
<td>90-th Percentile</td>
<td>6.79E-01</td>
<td>1.07E-02</td>
</tr>
<tr>
<td>95-th Percentile</td>
<td>8.83E-01</td>
<td>3.07E-02</td>
</tr>
</tbody>
</table>

4. Conclusions

The confidence building is a process that needs to be followed through all stages of the safety assessment of near surface disposal facilities. The results of sensitivity and uncertainty analyses related to the input parameters were presented. To enhance the confidence in the safety case we used a methodology compatible with the international experience. To increase the confidence in the results and to demonstrate compliance with regulatory requirements we employed different assessment techniques in a complementary manner. Thus, deterministic versus probabilistic models and conservative versus realistic data have been utilized. The deterministic approach has the advantage of implementation simplicity. A limitation of the deterministic approach is that there is often no systematic or complete coverage of the uncertainty space in parameter values. The main advantage of the probabilistic approach is that the model output is derived from a large number of input parameter sets. In this study we preferred a high degree of conservatism in scenario, mathematical model and data with the aim of demonstrating compliance with regulatory requirements. To investigate the safety margin included in the conservative estimates a comparison of best-estimate evaluations with conservative evaluations was performed. In this study best-estimate data were represented by infiltration flux, conceptual model of unsaturated zone (based on the solution of the inverse problem) and distribution coefficients of unsaturated zones.

REFERENCES

Regulatory and organizational framework for safe management and disposal of radioactive waste in Bangladesh

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Abstract. Radioactive waste (RW) in Bangladesh is generated from various applications. The wastes are generated from research and nuclear applications mainly in medicine, agriculture, industry and research. As with all radiation sources, radioactive waste is potentially hazardous to health and must therefore be managed in order to protect humans and the environment. All these RW needs to be safely managed at all stages up-to and including the ultimate disposal as per international and national standards. Regulation of the safe management and disposal of radioactive wastes in Bangladesh is the responsibility of the Nuclear Regulatory Authority. The Nuclear Safety and Radiation Control (NSRC) Act (Act No. XXI of 1993), in force since 22 July 1993 by approval of the National Parliament, assigns to Bangladesh Atomic Energy Commission (BAEC) to regulate and to inspect nuclear activities regarding radiological and nuclear safety, physical protection of nuclear materials, in order to verify that such materials are used exclusively for peaceful purposes. For the fulfilment of these functions the Act grants the BAEC the necessary legal competence to develop, to establish and to apply a regulatory system to all nuclear activities carried out in Bangladesh. The legal and regulatory framework employed to achieve safe management and disposal of radioactive wastes is quite detailed in the NSRC Act and NSRC Rules-97. Radioactive waste management activities are carried out in a facility at Central Radioactive Waste Storage and Processing Facility (CWSPF), Atomic Energy Research Establishment (AERE), Savar. This facility has been assigned to take all low-level radioactive wastes generated by nuclear applications in Bangladesh. Strategic guidelines for the management of the national nuclear activities and in particular for the radioactive waste management and decommissioning of the nuclear installations are provided in the NSRC Rules-97. Bangladesh is planning to sign the Joint Convention on the Safety of the Spent Fuel Management and on the Safety of the Radioactive Waste Management. This paper describes briefly the regulatory and organizational framework in which the management of radioactive wastes is carried out in Bangladesh. It also presents the criteria developed by the Nuclear Regulatory Authority (BAEC) to assess the long-term safety of disposal systems for radioactive wastes. The current status and future direction of the legal and regulatory framework for management and disposal of radioactive waste in Bangladesh is described.

1. Introduction

Bangladesh has no nuclear power reactors but the Bangladesh Atomic Energy Commission (BAEC) operates a 3 MW TRIGA Mark II research reactor (RR) and other nuclear facilities at its Atomic Energy Research Establishment (AERE) at Savar. The main sources of radioactive wastes are: use of sealed and unsealed radiation sources in medicine, industry, research, agriculture, etc. and operation & maintenance of the nuclear facilities (RR, Radioisotope Production Laboratories, Neutron Generator, Industrial & Research Irradiators, etc.). Radioactive wastes generated in Bangladesh are mostly low level radioactive waste, involving both short-lived and long-lived radionuclides. In general, this includes radioactive materials, which are no longer useful and have their origin from practice or intervention both with unsealed and sealed sources. Small quantities of radioactive waste including spent sealed radiation sources (SRS) are generated in Bangladesh. Much of this waste is short-lived low-level waste; and can be managed by delay-and-decay storage. Primary management of the short-lived low-level radioactive wastes is the responsibility of the generators. However, the BAEC's Radiation Monitoring and Waste Management Unit at AERE (Savar) has been conducting the collection, processing as well as storage of radioactive wastes including spent radiation sources (SRS). A facility, the Central Radioactive Waste Storage and Processing Facility, (CWPSF) has been
established in AERE. It is planned to relocate sealed sources (SRS) and radioactive wastes from hospitals, clinics, industries and research laboratories into this facility. Some spent sealed radiation sources (of hospitals, industries, research etc.) are collected and stored on-site and some are being safely transported and stored in shielded enclosures within AERE Savar. Currently, there is no facility for disposal of radioactive wastes in Bangladesh, but site investigation is in progress with the aim of having a pilot-scale near-surface waste repository for disposal of low and intermediate level waste (LIL) by 2010.

2. Institutional framework

In the framework of the Radioactive Waste Management in Bangladesh, the competent national bodies are the following:

2.1. **Bangladesh Atomic Energy Commission (BAEC)**

The BAEC under the Ministry of Science and ICT is the authority that issues the operating license for all nuclear and radioactive installations, after the positive technical advice of Nuclear Safety and Radiation Control Division (NSRCD). For installations related to radioactive waste storage and disposal, the concerted agreement of the Ministries of Environment and Health is also required.

2.2. **Nuclear Safety and Radiation Control Division (NSRCD)**

NSRCD is responsible for the regulation and supervision (by inspection) of nuclear installations in matters of nuclear safety and radiation protection. Any license granted by the BAEC incorporate the corresponding perceptive and legally binding report of NSRCD.

2.3. **National Nuclear Safety Committee (NNSC)**

This Committee is composed of experts from BAEC, NSRCD, and from various Ministries and Universities, and gives technical advice concerning the granting of licenses for nuclear installations.

2.4. **Radiation Monitoring and Waste Management Unit (RMWMU)**

All radioactive waste produced in Bangladesh is collected, segregated, conditioned and stored at CWSPF of RMWMU, AERE, Savar. Main components of the facility are listed below:

- Liquid waste is treated in chemical processing unit where precipitation is applied.
- Compactable solids are compressed in a compaction cell.

Spent sources are embedded into cement mortar with their original shielding. If the source activities are in several millicuries, sometimes dismantling is applied and segregated sources are conditioned in shielded drums.

3. Regulatory framework

3.1. **Legislation and regulation**

The Nuclear Safety and Radiation Control (NSRC) Act-93 (Act No. XXI of 1993)[1] and the NSRC Rules-97[2] are the legal basis for the control of ionizing radiation sources. Both the Act and Rules have been notified in Bangla, the national language. Authenticated English texts of the Act and Rules were also published. Moreover, the following Act/Rules have been drafted:


3.2. **Regulatory Infrastructure**

In Bangladesh, the relevant national authority for regulating activities involving radioactive sources is the Bangladesh Atomic Energy Commission (BAEC). The organogram of BAEC is shown in Fig. 1.
3.3. Waste Management Strategy

A strategy for radioactive waste management (RWM) would basically be a function of a national policy and such a policy has to be implemented through a legal framework. Sources of short half-lives are allowed to decay during temporary storage at user’s premises. Temporary storage is also beneficial to wait for sufficient numbers of depleted sources, so that all accumulated sources, similar in nature, can be sent for long-term storage or disposal in a single consignment. This helps in reduction of transportation cost. Different stages in the management system are: Temporary storage at user’s premises; Transport to waste management site; Treatment and conditioning; and Disposal. A general criterion is in force in Bangladesh for unrestricted release from any installation subject to either notification or authorization requirements. Radioactive materials from such practices can be unconditionally released from regulatory control if the radionuclides concerned comply with conditions regarding both activity concentration and radioactive half-life:

- activity concentration $1 \text{ Bq/g}$, and
- half-life $< 75 \text{ days}$.

In Bangladesh, there is not yet a LLW disposal facility and the radioactive waste from nuclear facilities is still stored at their points of origin. Radioactive waste from medical, industry and research activities is collected for temporary storage by CWSPF. NSRC Rules-97 provides performance objectives for a land disposal facility that releases to the general environment not result in an annual dose to an individual exceeding $0.25 \text{ mSv}$ to the whole body.

3.4. Guidance

The reference document concerning the radioactive waste management is the Technical Guide, issued by NSRCD, which provides waste classification as well as the technical requirements for the waste forms and the waste packages. Other relevant guidelines are provided with the Technical Guide “Quality Assurance Criteria”, where it is stated that the operator must submit to the regulatory authority a complete documentation concerning:

- Quality Assurance Programme,
- Adopted criteria for the waste conditioning facility design, operation and control,
- Results of product characterization.
4. Radioactive Waste Management System

4.1. Waste inventory
The overall national inventory of the radioactive waste, spent sources and spent fuel presently stored in different facilities in Bangladesh has been prepared by NSRCD in collaboration with CWSPF. The database is able to present the data in terms of volumes, mass, activity and physical status.

4.2. Waste classification
As established by the NSRC Rules-97, according to the radioisotopes characteristics and concentrations and having as principal reference possible options for final disposal, radioactive waste are classified into three Categories:

**Category I:** Waste which decay in a few months to radioactivity level below safety concerns (mainly hospital and research waste with $T_{1/2} < 1$ year).

**Category II:** Waste which decay to radioactivity level of few hundreds of Bq/g within few centuries. Activity of several radionuclides shall not exceed given values. Within Category II, two subcategories are defined:
- Solid waste whose activities concentration is below established limits, as listed in Table 1, which can be disposed of without further conditioning process;
- Waste with activity concentration above the established limits which need to be conditioned and must fulfill further requirements to be accepted for final disposal.

**Category III:** Long lived waste not included in category I and II; high level waste from reprocessing of spent fuel and alpha bearing waste from the fuel cycle and R&D activities.

<table>
<thead>
<tr>
<th>Radionuclides with $T_{1/2} &gt; 5$y</th>
<th>370 Bq/g</th>
<th>10 nCi/g</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{137}$Cs + $^{90}$Sr</td>
<td>740 Bq/g</td>
<td>20 nCi/g</td>
</tr>
<tr>
<td>Radionuclides with $T_{1/2} \leq 5$y</td>
<td>18.5 kBq/g</td>
<td>500 nCi/g</td>
</tr>
<tr>
<td>$^{60}$Co</td>
<td>18.5 kBq/g</td>
<td>500 nCi/g</td>
</tr>
</tbody>
</table>

4.3. Storage
Radioactive wastes are currently stored both at BAEC's facilities (especially, the AERE, AECD and NMCs) and at various users facilities. Most of the spent radiation sources are, presently, stored with appropriate shielding at their places of generation.

4.4. Management
In this paper radioactive waste is considered in two categories: as originated from unsealed sources or from sealed sources.

4.4.1. Management of unsealed sources
Unsealed radionuclides are utilised in human medicine for in vivo diagnosis, metabolic therapy and in vitro biological analysis. The most common types of radionuclides used in Bangladesh are $^{14}$C, $^{57}$Co, $^{51}$Cr, $^{59}$Fe, $^{67}$Ga, $^{123}$I, $^{125}$I, $^{131}$I, $^{32}$P, $^{89}$Sr, $^{90}$Sr, $^{99}$Tc, $^{201}$Tl, $^{133}$Xe, $^{90}$Y, etc. Solid radioactive waste consists mainly of protective clothing, plastic sheets and bags, gloves, masks, organs and tissues, animal carcasses, filters, overshoes, paper wipes, towels, metal and glass, hand tools, discarded radiopharmaceuticals containers and discarded equipment. It generally contains a relatively low level of radioactivity when compared to liquid wastes. Special consideration should always be given to the management of contaminated sharp objects, such as needles and syringes, scalpel blades, blood lancets, glass ampoules, etc. Short-lived solid radioactive wastes are stored in the waste storage rooms of the facilities until their activities reduce to an acceptable level to be released to the municipal waste disposal area. The liquid waste can be discharged to sewage system when its activity concentration comes down to permissible discharge level as per NSRC Rules-93.
4.4.2. Management of sealed sources

Sealed radiation sources are widely used in industry, medicine and research in Bangladesh. Sealed sources have a life cycle, which begins with manufacture and culminates in disposal. Each source life cycle comprises a number of potential stages. A source life cycle can involve individuals in the following key organisations: regulator, manufacturer, Original Equipment Manufacturer, distributor, user (one or subsequent users), waste management organisation, and operator of storage or disposal facility. The large number of organisations potentially involved and their interactions mean that life cycles tend to be complex and can be difficult to trace. The fundamental issue for which protection is required, is the prevention of over exposure of individuals or groups throughout the entire life cycle of sealed sources. Disused sealed sources which potentially represent medium and high radiological risks in Bangladesh are mainly Am-241, Ra-226, Kr-85, Co-60, Ir-192 and Cs-137. According to the NSRC Rules-97 all spent sources have to be sent to the manufacturer. However, the spent sources, which the manufacturer stopped its source related activities or the sources which were imported before the issue of the Rules-97, are stored in Central Radioactive Waste Storage and Processing Facility (CWSPF) of AERE.

4.5. Conditioning

The country's known spent Radium-226 sources of $t_{1/2}=1622\text{yrs} \ (~1\text{Ci} \approx 35.6 \text{ GBq})$ have been collected from different hospitals, industries, and conditioned & safely stored (temporary basis) within AERE campus (2000), following encapsulation of the sources (needles, applicators etc.) in stainless steel capsules, placement in heavy Pb-shielding device which has finally been placed & protected in 200 L capacity cement pre-fabricated MS drum under IAEA Model Project (INT/4/131): Sustainable Technologies for Managing Radioactive Wastes. On completion of Central Radioactive Waste Storage and Processing Facility (CWPSF), under construction in the AERE campus, the drums of the conditioned radium sources will be shifted & safely stored there.

5. Disposal facility

There is no existing facility for disposal of radioactive wastes in Bangladesh. However, a plan has been chalked out for development of a pilot-scale facility for disposal of LLW within the AERE by the year 2010. It is proposed that the pilot facility should be located in an area adjacent to the CWPSF to facilitate transportation and control of the facility. Geo-scientific investigation covering a 25 km region around the AERE campus for site selection for establishing a pilot-scale near-surface repository for short-lived LLW within AERE (by 2010) is in progress.

The radiological protection criteria applied by RA to the disposal of radioactive wastes establishes that no individual of the critical group shall be exposed to a risk higher than $10^{-5} \text{ y}^{-1}$ with optimization of the protection systems, and if the risk is not higher than $10^{-6} \text{ y}^{-1}$, the optimization requirement is not considered necessary. Those values are equivalent to a dose constraint of 0.3 mSv y$^{-1}$ and a dose reference level of 0.03 mSv y$^{-1}$ respectively. These criteria are consistent with the International Commission on Radiological Protection (ICRP) recommendations [3-4]. The main aims are to ensure that the individual risks are below the appropriate limits and to keep the radiological impact as low as reasonably achievable (ALARA).

6. Research and development programme

A research group has been formed to perform research activity for the development of a conceptual design of an engineered near surface LLW disposal facility, with a particular focus on the development of the safety assessment methodologies. In particular, and with the reference to a generic Near Surface Disposal Facility, the following standards have been approved or are still in process:

- Criteria for qualification of conditioned solid radioactive waste;
- LLW radiological characterization for near surface disposal;
- Waste package identification procedures;
- Packages and containers for LLW;
- Basic design criteria for an Engineering LLW disposal facility;
• Qualification criteria for the engineering barriers of a LLW disposal facility;
• Monitoring system for a LLW disposal facility.

7. Conclusion

The problem of management of radioactive waste in Bangladesh is not so serious at present because the wastes arising are small now. In terms of the magnitude of problems and the efforts already taken, the radioactive waste management is adequate for the present nuclear activities in Bangladesh. The legal and regulatory framework for management and disposal of radioactive wastes in Bangladesh is quite detailed. Waste prevention and minimisation is an essential element of any radioactive waste management strategy. The objective of waste minimisation is to reduce the activity and the volume of wastes for storage, treatment and disposal. The environmental impact will also be reduced, as well as the costs associated with contaminated material management. AERE is now in a position to establish a radioactive waste management facility and studies are now being carried out on the selection of the best place for the final storage of processed radioactive wastes. Research and development studies in AERE should continue in radioactive waste management with the aim of improving data, models, and concepts related to long-term safety of disposal of long-lived waste.

REFERENCES

Study of gas generation in drum L/ILW packages using hermetic containers

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Abstract. To obtain reliable estimates of the quantities and rates of the gas production in low and intermediate level radioactive waste (L/ILW) a series of measurements was carried out of drum waste packages produced and temporarily stored at the site of Paks Nuclear Power Plant (Paks NPP), Hungary. Ten drums filled with selected original L/ILW were placed into hermetic containers which were equipped with sampling valves for repeated sampling. Using the measured state parameters and the composition variation of the gas in the closed containers the gas generation rate in the stored L/ILW was calculated. Carbon, nitrogen and helium stable-isotope measurements and tritium and radiocarbon activity measurements were executed from the headspace gas samples.

1. Introduction

The operational low and intermediate level radioactive waste (L/ILW) produced in a nuclear power plant contains only very small amounts of long-lived radionuclides. Although it will decay to harmless levels in a relatively short time it needs to be disposed in a repository. During the storage of L/ILW significant quantities of gas may be produced. The main accountable processes for gas generation are principally the metal corrosion and microbial degradation of organic, particularly cellulose wastes. It is likely that a small proportion of the generated gas will be radioactive as a result of the incorporation of the isotopes $^3$H and $^{14}$C that are present within the waste.

In order to assess the implications of gas generation for the safety of a repository of L/ILW, it is important to gain an understanding of the principal mechanisms of gas generation and estimate the rate of gas formation and the possible radioactivity of generated gases. Preliminary estimates indicated that in L/ILW substantial quantities of gas would be produced in reactions involving certain components of the waste forms and their containers. It was concluded that gases, which could be mainly produced, are: hydrogen (by corrosion of metals), methane and carbon dioxide (by microbial activities) [1-6]. Because of the relatively low radionuclide content the contribution of radiolysis to the gas generation is not significant. Radioactive gases had to be considered are mainly the gases noted above, in which atoms of hydrogen and carbon are replaced by the $\beta$-emitters tritium ($^3$H) and radiocarbon ($^{14}$C), respectively [7]. In spite of the wide-range experimental investigations concerning the gas generation (Nirex, Pacific Northwest Laboratory, Westinghouse Hanford Company, Argonne National Laboratory), the available data measured in real L/ILW drums are very limited [8].

To obtain reliable estimates of the quantities and rates of the gas production in L/ILW a series of measurements was carried out of waste packages produced and temporarily stored at the site of Paks Nuclear Power Plant (Paks NPP), Hungary. In Paks NPP these wastes are packed into special drums of steel, which are not closed hermetically for gases. Preliminary results showed that significant gas production is detectable in these drum wastes, but the estimation of the gas generation rates is complicated because of the uncontrolled gas-transport between the headspace gas of the drums and the outer air [9-10].

This paper presents the results of a long-term measurement series using hermetic containers to make more precise quantitative estimation of the gas generation rate in the case of the drum L/ILW of the Paks NPP.
2. Experiments and materials

Ten drums filled with selected original L/ILW were placed into hermetic containers equipped with sampling valves for repeated sampling (Fig. 1). These hermetic containers were stored at the same site where the L/ILW is stored primarily in the Paks NPP.

FIG. 1: Hermetic containers designed for drum L/ILW gas generation measurements (DV,CV: gas sampling valves for the drum and the container; P,T: pressure and temperature sensors; P/T MS: pressure and temperature monitoring system; M: manometer; SV: safety-valve).

The pressure and the temperature of the headspace gas in the containers were monitored continuously. Qualitative gas component analyses of headspace gases of drums and their containers were executed regularly by quadruple mass spectrometer. The gas generation rate in the stored L/ILW was calculated by the measured state parameters and the composition variation of the gas in the closed containers.

Samples from the headspace gases were delivered to the Laboratory of Environmental Studies of the Institute of the Nuclear Research of the Hungarian Academy of Sciences (ATOMKI) for further isotope-analytical measurements. Stable isotope measurements were executed from the CO₂, CH₄ and N₂ fractions by stable isotope ratio mass spectrometer (ATOMKI). Noble gas (He) measurements were done by noble gas mass spectrometer (Fisons VG5400). The tritium content of the vapour, H₂ and CH₄ fractions was measured in H₂O chemical form by a low background liquid scintillation counter (Canberra Packard TRICARB 3170TR/SL). The ¹⁴C content of the CO₂ and CH₄ fractions of the headspace gas samples was measured by a low background gas proportional counter system (ATOMKI).

3. Results and discussion

It was clearly indicated that the gas generation rate is relatively high in the L/ILW drums independently the chemical type of the main components of the stored waste. It means that high volume of generated gases must be considered during the storage. Our results showed that the main generated gases in L/ILW are carbon-dioxide, methane, hydrogen and nitrogen. The typical rates were 0.05-0.2 STP litre gas/day for CO₂ and CH₄ generation, and less than 0.02 STP litre gas/day for H₂. Because of the typical vanishing of the O₂ from the headspace gases in the first few months no explosive gas mixture was indicated in the L/ILW drums during the investigated storage period.

The stable carbon isotope measurements show that the main source of the CO₂ gas is the degradation of organic matter in the waste. The low ¹³C content indicates microbial degradation processes as the main sources of CH₄ in the headspace gas. The He isotope ratios represent ³He enrichment in the headspace gases produced by tritium decay in the waste.
No relationship was found between the total activity of the waste stored in the drums and the amount and rate of the gas generation. It seems to be likely that the gas formation is controlled by corrosion and/or bacterial activity and the radiolysis plays a minor role.

Tritium activity concentration values show high diversity. Significant variation of tritium content in individual drums with time was also typical. The maximal value was more than 20 Bq/litre. The typical tritium activity concentrations were between 0.1 and 10 Bq/litre. The main tritium content always was related to the vapour fraction, but when the methane became main component in the headspace gas it also could contain significant amount of $^3$H.

The radioactivity of the carbon in the gas phase of the L/ILW drums are always significantly higher (10 or 100 times) than in the air. This $^{14}$C enrichment also appears in the methane fraction if it became main component in the headspace gas. Maximal measured radiocarbon activity concentration of the bulk gas in drum L/ILW packages was about 3.0 Bq/litre. Typical $^{14}$C activity values of the headspace gases were between 0.1 and 2.0 Bq/litre.

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Safety assessment of a large diameter well type repository

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Abstract. In 1998, the Moscow enterprise “Radon” constructed a demonstration site for large diameter well type (LDW) repositories. The well type repositories were designed for low and intermediate level waste storage and disposal. Such repositories could be sources of environmental contamination with radionuclides, which are transferred outside engineered barriers. Specialists of the Scientific and Engineering Centre for Nuclear and Radiation Safety (SEC NRS) have carried out a safety assessment of the LDW-1 disposal facility using AMBER software. The normal evolution and alternative scenarios were developed. An alternative scenario considered a presence of ground aquifer which crossed an LDW disposal facility.

1. Introduction
The management of low- and intermediate-level radioactive waste is carried out in Russia by the “Radon” system, that was established in the beginning of the 1960s in the Soviet Union and included 35 enterprises. There are now 16 specialized enterprises of this system located in various regions of the Russian Federation [1]. They were designed for transportation, preliminary processing and storage of solid, liquid waste as well as spent sealed sources.

“Radon” specializes in radioactive waste management in scientific institutions, industry, agriculture and medical institutions, as well as the aftermath of the Chernobyl accident.

In 1998 Moscow enterprise “Radon” constructed a demonstration site for large diameter well type (LDW) storage facilities [2]. Design of the first constructed LDW storage facility (LDW-1x) allowed retrieval of stored containers [2].

The design of the LDW-1 facility (Fig. 1), which was constructed a year ago, does not provide retrieval of containers from the facility – cavities are filled up with a bentonite-cement mixture, which after strengthening becomes one of the engineered barriers [2].

![FIG. 1: Large diameter well (LDW-1)](image)

Wells for the LDW-1x and LDW-1 facilities contain metallic bottom-covered casing pipes [2], which have the same design for both facilities.

Specialists of the Scientific and Engineering Centre for Nuclear and Radiation Safety (SEC NRS) have carried out safety assessment of LDW-1 disposal facility.

The safety assessment of the LDW-1 disposal facility was carried out using a multi-step process involving [3]:

- assessment context;
- description of disposal system;
- development and justification of scenarios;
- formulation and implementation of models;
- analysis of results.

2. Description of the disposal system

LDW-1 disposal facility is buried under 40 meters of earth, inner diameter equals 1.5 m. Volume in the well for radioactive wastes is 19.2 m³. The radioactive wastes that must be disposed of in the LDW-1 disposal facility are conditioned in steel drums [4].

The LDW-1 disposal facility realizes a multi-barrier principle. The following barriers are used [4]:

- waste forms;
- steel drums;
- steel casing pipe;
- bentonite-cement mixture that fills up free space in the facility and makes a barrier in the bottom and side parts of the facility;
- host rock (clays, clay loam).

As for the geosphere, a host rock in a place of the demonstration site is characterized by the presence of sand lenses [2]. One of the lenses was revealed during a drilling of a well for the LDW-1x storage facility.

For safety assessment of the LDW-1 disposal facility the significant radionuclides which are present in L/ILW radioactive wastes had been chosen (Table 1):

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Permissible specific activity in radioactive wastes disposed of in near-surface disposal facilities [5].</th>
<th>Permissible activity in LDW-1 disposal facility *, Bq kg⁻¹</th>
</tr>
</thead>
<tbody>
<tr>
<td>¹³⁷Cs</td>
<td>1.7·10¹⁴ Bq/m³</td>
<td>3.3·10¹⁵</td>
</tr>
<tr>
<td>⁹⁰Sr</td>
<td>2.6·10¹⁴ Bq/m³</td>
<td>5.0·10¹³</td>
</tr>
<tr>
<td>²³⁵U</td>
<td>3.7·10⁶ Bq/kg</td>
<td>1.4·10¹¹ **</td>
</tr>
<tr>
<td>²³⁹Pu</td>
<td>3.7·10⁶ Bq/kg</td>
<td>1.4·10¹¹ **</td>
</tr>
</tbody>
</table>

* Volume of wastes 19.2 m³ [4].
** Density of cemented wastes 2000 kg/m³.

3. Development and justification of scenarios

Scenarios are descriptions of alternative, but internally consistent, future evolutions and conditions of the waste disposal system. They handle future uncertainty directly by describing alternative futures and allow for a mixture of quantitative analysis and qualitative judgements. Essentially, the main purpose of scenario generation in the post-closure safety assessment of a radioactive waste disposal system is therefore to use scientifically-informed expert judgement to guide the development of descriptions of the disposal system and its future behaviour [3].
Foundation of the LDW-1 disposal facility (bentonite-cement layer ~ 2 m) and a host rock (clays) prevent an aquifer, located below the base of the facility, from radionuclides infiltration. Radionuclides transfer from the facility to the geosphere is defined by diffusion process, which will not lead to considerable contamination of the aquifer (normal evolution scenario).

Since the host rock in a place of the demonstration site is characterized by the presence of sand lenses, it was assumed that a lens, which was revealed during a drilling of a well for the LDW-1x storage facility, is being a ground aquifer, located above the base of the LDW-1 facility. Also was also assumed that in 300 years after LDW-1 closure the bentonite-cement walls in the depth of the ground aquifer have degraded as a result of a violation of technology of bentonite-cement barriers construction. Cementitious waste forms have degraded as well. Radionuclides leached from waste forms infiltrate through the degraded barriers to the ground aquifer (alternative scenario). According to recommendations of the IAEA experts [6] concentrations of radionuclides in groundwater 100 m out from the LDW-1 disposal facility were calculated.

4. Formulation and implementation of models

A computer code AMBER was used to calculate migration of radionuclides from the LDW-1 disposal facility. AMBER is a flexible software tool that allows the user to build their own dynamic compartmental models to represent the migration and fate of contaminants in a system, for example in the surface and sub-surface environment. Contaminants in solid, liquid and gaseous phases can be considered [7]. The mathematical representation of the intercompartmental transfer processes takes the form of a matrix of transfer coefficients that allow the compartmental amounts to be represented as a set of first order linear differential equations. For the \( i^{th} \) compartment, the rate at which the compartment inventory changes with time is given by:

\[
\frac{dN_i}{dt} = \left( \sum_{j \neq i} \lambda_{ji} N_j + \lambda_{Ni} M_i + S_i(t) \right) - \left( \sum_{j \neq i} \lambda_{ij} N_i + \lambda_{jN} N_j \right)
\]

where

\( i, j \) indicate compartments;

\( N_i, M_i \) are the amounts (Bq) of radionuclides \( N \) and \( M \) in a compartment (\( M \) is the precursor of \( N \) in a decay chain);

\( S_i(t) \) is a time dependent external source of radionuclide \( N \), Bq y\(^{-1}\);

\( \lambda_i, \lambda_N \) is the decay constant for radionuclide \( N \) (in y\(^{-1}\)); and

\( \lambda_{ji}, \lambda_{ij} \) are transfer coefficients (y\(^{-1}\)) representing the gain and loss of radionuclide \( N \) from compartments \( i \) and \( j \).

5. Analysis of results

The results of modelling for the normal evolution scenario did not show the exceedance of permissible activity levels IL\(_{\text{water}}\) (intervention levels [7]) for all radionuclides in aquifer 100 m out from LDW-1 disposal facility.

Calculations for alternative scenario indicated the exceedance of intervention levels for Pu-239 (IL\(_{\text{water}}\) = 0.56 Bq/kg) and U-235 (IL\(_{\text{water}}\) = 3 Bq/kg) in ground aquifer 100 m out from LDW-1 disposal facility. The exceeding of intervention level for Pu-239 is observed in 31000 years after closure of the facility, for U-235, after 20 000 years (Fig. 2). Thus, ingestion of water for drinking purposes from water well located 100 m out from LDW-1 disposal facility becomes unallowable in 20 000 years after closure of the facility. There is no danger from short-lived radionuclides Cs-137 (IL\(_{\text{water}}\) = 11 Bq/kg) and Sr-90 (IL\(_{\text{water}}\) = 5 Bq/kg) while drinking water from water well located 100 m out from LDW-1 disposal facility.

6. Conclusion

The results obtained for the normal evolution scenario of the LDW-1 disposal facility suggest that doses should be low and within acceptable limits. However, the results for the alternative scenario indicate the possible exceedance of permissible activity levels in ground aquifer for long-lived radionuclides.
Therefore, further work is necessary to show that such alternative scenario is not realistic for large diameter well type repositories for radioactive waste disposal. Such work could include taking into account in fuller detail:

- composition of radioactive wastes (especially inventory of long-lived radionuclides);
- sufficiency of the engineered barrier system;
- site specific features for disposal sites (especially hydrogeology).

It is also necessary to develop criteria specifically relevant to radioactive waste disposal facilities, which specifically address long issues and low-probability/high consequence events, taking into account international guidance and recommendations as they may be appropriately applied within the Russian Federation.

REFERENCES


A safety reassessment of “La Piedrera” disposal facility

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Abstract. The purpose of this paper is to present a very simple and conservative safety reassessment of the La Piedrera radioactive waste disposal facility, which confirms the low radiological impact on the environment and the public; making use of the methodology developed by the IAEA under the ISAM project [3]; executing the calculations with the AMBER code and taking advantage of more advanced calculation tools than used back in 1984.

1. Introduction
The La Piedrera radioactive waste disposal facility was build in 1984. The purpose of this facility was to dispose off the radioactive waste originated as a consequence of the remediation actions for the 1984 accident in Ciudad Juárez, Chihuahua, which involved the melt down of a Co-60 teletherapy source at a foundry facility. Considerable amounts of contaminated soil, metal pieces, and steel bars for construction had resulted from that accident.

2. Evaluation context
The assessment will be limited to those waste forms that are susceptible to be affected in the short term by the rainwater entering to an early fractured system and to those processes which can cause a more immediate radiological impact, these considerations were taken due to the relatively short half life of the only contaminant (Co-60, T_{1/2} = 5.27 years) and to the fact that the facility’s Institutional Control (40 years), will cover more than seven half lives of Co-60. The end point of this assessment is to demonstrate compliance with dose limits for public, considering the individual dose to a member of the critical group.

3. System description
The “La Piedrera” radioactive waste disposal facility is located at Samalayuca, 55 km south of Ciudad Juárez, Chihuahua.

3.1. Near field
The disposal system consists of 8 vaults with concrete barrier plus one without concrete barrier; the dimensions of vaults are shown in Figure 1. A cover of clean native soil was deposited over the waste.
The only contaminant radionuclide in “La Piedrera” is Co-60, which has a half life of 5.26 and decays to Ni-60 which is a stable isotope. The total activity of Co-60 at the time of its disposal was 443 Ci; however not all this activity was considered in the assessment as can be seen in Table I. Figure 2 shows the construction stage of a vault.

Table I. Forms and quantities of radioactive waste disposed in La Piedrera [1]

<table>
<thead>
<tr>
<th>FORMS OF DISPOSED WASTE</th>
<th>ACTIVITY ENE 1984 (Ci)</th>
<th>MASS (Tons)</th>
<th>VOLUME (m³)</th>
<th>COMMENTS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Contaminant embedded in steel bars for construction</td>
<td>124</td>
<td>2930</td>
<td>978</td>
<td>Forms considered in the assessment. Material likely to be leached and transported by advection</td>
</tr>
<tr>
<td>Metallic pieces for furniture</td>
<td>9</td>
<td>200</td>
<td>700</td>
<td></td>
</tr>
<tr>
<td>Scrap metal, without conditioning.</td>
<td>2</td>
<td>1 950</td>
<td>2 445</td>
<td></td>
</tr>
<tr>
<td>Metal in process</td>
<td>74</td>
<td>1 738</td>
<td>434</td>
<td></td>
</tr>
<tr>
<td>Co-60 distributed in contaminated soil, without conditioning.</td>
<td>12</td>
<td>29 181</td>
<td>16 212</td>
<td>Form considered in the assessment. Transport of contaminant by water flow</td>
</tr>
<tr>
<td>Pellets contained in 200 L drums reinforced with concrete.</td>
<td>158</td>
<td>860 (Drums)</td>
<td>179</td>
<td>Forms not considered in the assessment, it is assumed that they will stay intact for more than 40 years</td>
</tr>
<tr>
<td>Contaminant in the back of a pick up vehicle, covered with 0.5 m of concrete</td>
<td>60</td>
<td>N/A</td>
<td>7.0</td>
<td></td>
</tr>
<tr>
<td>Cobalt in the original source container</td>
<td>5</td>
<td>N/A</td>
<td>0.5</td>
<td></td>
</tr>
</tbody>
</table>

Due to the half life of Co-60, only the waste forms that are suitable for contaminant release via leach and advection processes were considered as indicated in Table I.

3.2. Geosphere

The fractic level is 14 m depth from the surface in its closest level (rain season), permeability ranges, between 1E-3 and 6E-3 cm/sec, permeability coefficient 3.5E-4 a 5E-4 cm/s, Soil porosity 38% to 41%, density of solids 2.63 g/cm³, volumetric weight 1.62 ton/ m³, hydraulic conductivity 0.017 m/s, Data obtained from [2], the fenced area of the facility is 3.0E4 m², the assumed volume of aquifer is 1.0E4 m³.

3.3. Weather

The facility is located in an arid zone with average annual temperatures of 17.6 °C, with average annual precipitations of 270.1 mm in the rain season (summer), the mean annual evapotranspiration is 250 mm. Data obtained from [2].

3.4. Biosphere

Biosphere data are not required in this assessment, owing to practically zero consumption of native fauna and flora. Additionally there is very low agricultural and cattle activities in the region; and this condition should not change within the next 20 years.

4. Scenario

Instead of going through a FEP’s screening process, an altered evolution scenario was considered for this assessment, which is based on the half life of Co-60, the length of institutional control period, the characteristics of the site and on the human activities in the region. Scenario characteristics are shown in Table II.

Due to the fact that the site has almost non-existent agricultural and cattle activities, exposure paths such as ingestion of crops and animal products were not considered.
Table II: Scenario characteristics

<table>
<thead>
<tr>
<th>TYPE</th>
<th>SCENARIO</th>
</tr>
</thead>
<tbody>
<tr>
<td>Altered Evolution</td>
<td>Early General failure of the concrete barrier (earthquake), container no longer acts as a barrier to water</td>
</tr>
<tr>
<td></td>
<td>Water reaches the waste, lixiviation, and advection occurs</td>
</tr>
<tr>
<td></td>
<td>Aquifer contamination</td>
</tr>
<tr>
<td></td>
<td>Paths of exposure to human: inhalation and unfiltered drinking water</td>
</tr>
</tbody>
</table>

5. **Formulation and implementation of models**

Advective flow from compartment $i$ to compartment $j$ is given by:

$$\Phi = \theta_i \nu_i C_i \quad \text{or} \quad \Phi = \lambda_{ij} \theta_i R_i C_i V_i$$

Where $\chi_{ij}$ is the common area between compartments $i$ and $j$, $C_i$ is the concentration of the radionuclide in donor compartment $i$, $\theta_i$ is the cinematic porosity of donor compartment $i$, $\nu_i$ is the advective velocity, $R$ is the retardation factor of the radionuclide in donor compartment $i$, $V_i$ is the volume of donor compartment. Combining the two equations

$$\lambda_{ij} = \frac{(\nu_i \chi_{ij})}{(R_i V_i)} \quad \text{or} \quad \lambda_{ij} = \frac{\nu_i}{(R_i L_i)}$$

Where $L_i$ is the length of donor compartment $i$ in the direction of advective flow.


A scheme of the model implemented in the AMBER code is shown in figure 3. Descriptions of the compartments defined are given in Table III. Inhalation and drinking water processes are not modelled in AMBER, they were drawn for illustrative purposes, and same applies for the Human compartment.

**FIG. 3: Model scheme in AMBER.**

Table III: Description of compartments

<table>
<thead>
<tr>
<th>COMPARTMENT</th>
<th>DESCRIPTION</th>
</tr>
</thead>
<tbody>
<tr>
<td>Radwaste</td>
<td>Radioactive waste in metallic form</td>
</tr>
<tr>
<td>Inner soil</td>
<td>Soil around the metallic waste</td>
</tr>
<tr>
<td>Outer soil</td>
<td>Soil out of the vault</td>
</tr>
<tr>
<td>Surface</td>
<td>Soil in the surface of the facility (within the fenced area)</td>
</tr>
<tr>
<td>Well</td>
<td>Aquifer from which drinking water is abstracted</td>
</tr>
</tbody>
</table>
6. Calculation and results

In order to make a conservative calculation, only forward transfer for advection and dispersion processes was considered, credit was not given to the concrete barrier, and the activity is considered uniformly distributed in the vaults.

FIG. 4: Activity in well compartment

FIG. 5: Activity in surface compartment

Figures 4 and 5 show the activity in the Well and Surface compartments respectively; considering the maximum values of activity for each compartment, the volume of aquifer and the fenced area, maximum concentrations of 100 Bq/m³ for the Surface compartment and 150 Bq/m³ for the Well compartment are obtained.

With these concentrations, and using the models of [4], individual doses via inhalation and ingestion to a member of the critical group can be obtained considering the following:

- a) ingestion rate of unfiltered water from the well, 0.7 m³/y,
- b) inhalation rate of air, 1.8 m³/h,
- c) time spent by an intruder in the Surface compartment, 730 h/y (daily walk through the site),
- d) total suspension of the surface activity contained in 1 m² into an air column 3.0 m high.

Resulting doses are shown in Table IV. The sum of the two values is considerably lower than both, the Mexican dose limit for public of 5.0 mSv per year and the IAEA’s dose limit [5] of 1 mSv per year.

Table IV: Resulting doses

<table>
<thead>
<tr>
<th>Medium</th>
<th>Concentration</th>
<th>Inhalation/Ingestion rate</th>
<th>Dose coefficient [5] (Sv Bq⁻¹)</th>
<th>Committed effective dose per year</th>
</tr>
</thead>
<tbody>
<tr>
<td>Air above Surface</td>
<td>33.3 Bq/m³</td>
<td>1314 m³/y (air)</td>
<td>3.1E-8</td>
<td>0.356 mSv/y</td>
</tr>
<tr>
<td>Water</td>
<td>150 Bq/m³</td>
<td>0.7 m³/y (water)</td>
<td>3.4E-9</td>
<td>0.357 µSv/y</td>
</tr>
</tbody>
</table>

REFERENCES


Uncertainties of radionuclide migration parameter values obtained from in-situ tracer experiments

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Abstract. One of the key issues in safety assessment of high-level nuclear waste disposal is evaluating the effects of uncertainty inherent in radionuclide migration parameter values. In this paper, radionuclide transport parameter values and error variances (uncertainties) from in-situ tracer experiments, carried out in a single fracture at the Äspö Hard Rock Laboratory (HRL) in Sweden, are identified by solving the inverse problem in a framework of the maximum likelihood theory. From the results, it is found that the parameter value uncertainty caused by a conceptual model of radionuclide migration is greater than that caused by a fluctuation in the observed breakthrough curve data.

1. Introduction

Fracture, advection-dispersion, sorption on fracture surfaces, diffusion into or out of immobile rock matrix, diffusion and sorption in the rock matrix, and radioactive chain decay are considered while evaluating the extent of radionuclide migration for a safety assessment of high-level nuclear waste disposal. Since the number of migration parameters is quite large, it is crucial to establish the procedure for identifying parameter values based on laboratory and field experiments and to evaluate the uncertainties of these identified parameter values. In this paper, a parameter identification methodology from in-situ tracer experiments is presented and the uncertainties of identified parameter values are presented.

2. Methodology

The governing equations of radionuclide migration in a fracture and in an immobile rock matrix \( m \) are written as described in Ref. [1]:

\[
\begin{align*}
R \frac{\partial C}{\partial t} + q \nabla \nabla \nabla C - \frac{\partial}{\partial x} D_L \frac{\partial C}{\partial x} + R \lambda C - \frac{1}{b} D_e \frac{\partial C_m}{\partial \omega} & = 0 \\
R_m \frac{\partial C_m}{\partial t} - \frac{\partial}{\partial \omega} D_m \frac{\partial C_m}{\partial \omega} + R_m \lambda C_m & = 0
\end{align*}
\]

where \( q \) is true velocity \([\text{m/s}]\), \( D_L \) is longitudinal dispersion coefficient \([\text{m}^2/\text{s}]\), \( \alpha_L \) is longitudinal dispersivity \([\text{m}]\), \( D_e \) is free-solution diffusion coefficient \([\text{m}^2/\text{s}]\), \( \lambda \) is the decay constant \([1/\text{s}]\) (= ln2/\( T \)), \( T \) is half-life period \([\text{s}]\), \( b \) is half fracture aperture \([\text{m}]\), \( D_m \) is effective diffusion coefficient of rock matrix \([\text{m}^2/\text{s}]\), \( \theta_m \) is porosity of rock matrix \([-]\), \( D_m \) is pore-water diffusion coefficient of rock matrix \([\text{m}^2/\text{s}]\), \( C, C_m \) are concentrations in fracture and rock matrix, respectively \([\text{mol/m}^3]\), \( \omega \) is distance perpendicular to the fracture surface \([\text{m}]\) (0 ≤ \( \omega \) ≤ \( d \)), \( d \) is matrix diffusion depth \([\text{m}]\), \( t \) is elapsed time \([\text{s}]\). We assume instantaneous, reversible and linear sorption, and thus the retardation factors are given by:

\[
R = 1 + \frac{K_m}{b}, \quad R_m = 1 + \frac{\rho_m K_m}{\theta_m}
\]

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where \( K_a \) is the distribution coefficient on fracture surfaces \([m^2/kg]\), \( K_d \) is distribution coefficient of rock matrix \([m^3/kg]\) and \( \rho_m \) is dry density of rock matrix \([kg/m^3]\).

In this study, the optimum parameter values and error variances are identified from tracer breakthrough curves by solving the inverse problem in a framework of the maximum likelihood theory. The log-likelihood criterion \( S \) is written as described in Ref. [2]:

\[
S = \frac{U}{\sigma^2} + N \ln|V| + N \ln\sigma^2 + N \ln(2\pi) - U = \sum_{i=1}^{N} \left( C_i - C_i^* \right)^T V^{-1} \left( C_i - C_i^* \right) \tag{4}
\]

where \( U \) is the concentration residual criterion \([-]\), \( \sigma^2 \) is error variance \([-]\), \( N \) is total number of concentration measurements \([-]\), \( V \) is symmetric positive definite matrix, \( C_i \), \( C_i^* \) is observed and calculated concentrations, respectively. Akaike’s information criteria (AIC) written as follows [3] is used for identifying the best model among alternatives.

\[
AIC = S + 2N \tag{5}
\]

where \( N \) is the number of identified parameters.

3. Tracer migration experiments at the Äspö HRL [4]

In order to understand tracer retention mechanism in a fractured rock, the tracer migration experiments were conducted in a single fracture, Feature A, located close to the drift at depth of 400 meters, in the Äspö HRL. Feature A is interpreted as a single, steeply dipping, NW trending fault plane (N29W/79E) characterized as a reactivated shear structure. The transmissivities of Feature A obtained from in-situ permeability tests range between \(1.2 \times 10^{-8}\) and \(3.6 \times 10^{-7}\) \(m^2/s\). The radially convergent tracer migration experiments were conducted between two boreholes, a distance of 4.68 meters apart. Tracers were injected into one borehole without pressure and were pumped from the other borehole at a pumping rate of 200 cc/min. The conservative (non-sorbing) tracers, uranine and HTO (half-life 12.3y), and sorbing tracers, \(^{22}\text{Na}\) (2.6d), \(^{85}\text{Sr}\) (65d), \(^{133}\text{Ba}\) (10.4y), \(^{86}\text{Rb}\) (18.7d) and \(^{134}\text{Cs}\) (2.1y), were used. The tracer breakthrough curves of HTO and \(^{22}\text{Na}\), obtained from the experiments, exhibit double peaks as can be seen in Figure 1. This implies that there are two different, dominant pathways between the two boreholes.

4. Analytical conditions

Since it is assumed that water flows as a one-dimensional channel in a fracture plane as noted in Ref. [5], a one-dimensional model is used for the analysis. In addition to the one-pathway model, two- and three-pathway models are used in order to better simulate the aforementioned double peaks of breakthrough curves. Among a number of the radionuclide parameters described in Equations (1) and (2), porosity, dry density, and an effective diffusion coefficient can all be accurately measured through laboratory tests and are fixed as shown in Tables I and II. The matrix diffusion depth is also fixed at 0.05m, which is the width of alteration along Feature A. The initial values of the distribution coefficient, true velocity, dispersivity and aperture are determined as shown in Tables I and II. In the analysis, true velocity, dispersivity and aperture are identified from non-sorbing HTO data and the distribution coefficient of each sorbing tracer is identified from the corresponding tracer data.
Table I: Fixed and initial values for radionuclide

<table>
<thead>
<tr>
<th>Tracer</th>
<th>Diffusion coef. $[\times 10^{-6} \text{m}^2/\text{h}]$</th>
<th>Distribution coef. $[\times 10^{-5} \text{m}^3/\text{kg}]$</th>
</tr>
</thead>
<tbody>
<tr>
<td>HTO</td>
<td>8.4</td>
<td>0</td>
</tr>
<tr>
<td>$^{22}\text{Na}$</td>
<td>4.8</td>
<td>0.14</td>
</tr>
<tr>
<td>$^{85}\text{Sr}$</td>
<td>2.8</td>
<td>0.47</td>
</tr>
<tr>
<td>$^{133}\text{Ba}$</td>
<td>3.0</td>
<td>20</td>
</tr>
<tr>
<td>$^{86}\text{Rb}$</td>
<td>7.3</td>
<td>40</td>
</tr>
<tr>
<td>$^{134}\text{Cs}$</td>
<td>7.3</td>
<td>600</td>
</tr>
</tbody>
</table>

Table II: Fixed and initial values for system

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Porosity [-]</td>
<td>0.004</td>
</tr>
<tr>
<td>Dry density $[\text{kg/m}^3]$</td>
<td>2,689</td>
</tr>
<tr>
<td>Pathway-A: True velocity [$\text{m/h}$]</td>
<td>0.94</td>
</tr>
<tr>
<td>Pathway-B: True velocity [$\text{m/h}$]</td>
<td>0.20</td>
</tr>
<tr>
<td>Pathway-C: True velocity [$\text{m/h}$]</td>
<td>0.57</td>
</tr>
<tr>
<td>Dispersivity [$\text{m}$]</td>
<td>0.47</td>
</tr>
<tr>
<td>Aperture [$\text{m}$]</td>
<td>$5.0 \times 10^{-3}$</td>
</tr>
</tbody>
</table>

5. **Uncertainties derived from in-situ experiments**

The observed and calculated breakthrough curves are plotted in Figure 3, and the 95-percentile confidence ellipses of identified true velocity and dispersivity is plotted on contours of log-likelihood criterion as shown in Figure 4. The calculated breakthrough curves from the two-pathway model consistent with the observed curves, while the one-pathway model does not yield a double peak of HTO and $^{22}\text{Na}$ as shown in Figure 3. In addition, the ellipse, as well as the AIC value, for the two-pathway model is smaller than those for the one-pathway model as shown in Figs 4 and 5. Therefore, it is noted that a two-pathway model is better than a one-pathway model. In contrast, two- and three-pathway models yield almost the same breakthrough curves and parameter values for pathways -A and -B as can be seen tabulated in Table III and AIC values as described in Figure 5. It is assumed that pathway-C of the three-pathway model, where the flux rate is smallest, does not have a great impact on the configuration of the breakthrough curves. In addition, the identified parameter values obtained from one- and two-pathway models range larger than the 95-percentile ellipse of each model. Therefore, it is noted that the uncertainty caused by various models is larger than that caused by data fluctuation.

**FIG. 3: Calculated breakthrough curves plotted on observed data**
6. Conclusions

In this study, radionuclide transport parameter values and their uncertainties are identified from in-situ tracer experiments by solving the inverse problem in a framework of the maximum likelihood theory. From the results, it is concluded that the parameter value uncertainty caused from the fluctuation of the observed tracer breakthrough curve is smaller than that caused from the model variation. It is also noted that model identification criteria, AIC, is quite effective for reducing model uncertainty.

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Approach to establishing safety margin for uncertainty in measurement and nuclide spectrum in clearance level inspection

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Abstract. In the monitoring for compliance with the clearance level, the concentrations of objective nuclides, such as alpha or low-energy beta emitters, can be estimated without direct gamma measurement by assuming the existence of objective nuclides at geometric mean concentrations or by using previously assessed information on nuclide spectra and measurement results for a key gamma nuclide. To determine whether clearance can be carried out, the uncertainty in the mean concentrations and concentration ratios to the key gamma nuclide should be appropriately considered, in addition to the measurement uncertainty. In this work, the concept of the clearance level has been reconsidered and a new approach to establishing an appropriate safety factor of the monitoring for compliance with the clearance level has been proposed. This approach was adopted in the draft of standard of “Monitoring for Compliance with Clearance Level” prepared by the Standards Committee (SC) of the Atomic Energy Society of Japan (AESJ).

1. Introduction

The International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (BSS) [1] specifies the basic international requirements to protect human health against exposure to ionizing radiation and for the safety of radiation sources. The BSS defines the concepts of exclusion, exemption and clearance. The applications of these concepts are given in the safety guide, RS-G-1.7 [2].

RS-G-1.7 provides us with activity concentration levels that are equivalent to clearance levels and also shows that the sum (hereinafter termed $\Sigma C/CL$) of the estimated nuclide concentrations (C) divided by each clearance level (CL) should be lower than 1 to satisfy the clearance criterion for a mixture of nuclides. It is also stated that the verification of the levels should be based on a procedure that may include direct measurement of materials, laboratory measurements of representative samples, and the use of properly derived radionuclide relationships.

As seen in RS-G-1.7, the estimation of target nuclide concentrations for clearance inspection is based on the direct measurement of the concentrations of readily monitored nuclides. However, for nuclides whose concentrations cannot be measured by gamma measurement, an estimation method using previously assessed information on nuclide spectra and measurement results for a key gamma nuclide (hereinafter termed the nuclide spectrum (NS) method) can be applied. For target nuclides unsuitable for the NS method, the determination of their mean concentrations may be adopted as a conservative means of clearance inspection (hereinafter termed the mean concentration (MC) method).

In Japan, in March 1999, the Nuclear Safety Commission, an advisory organization to the Japanese government, published a report on clearance levels for solid materials, e.g. concrete and metal, generated from the decommissioning and operation of the main Japanese reactors [3]. In July 2001, the basic concepts of clearance inspection were also published [4]. In this report, there is the requirement that the relationship $\Sigma C/CL<1$ should be confirmed in the case of clearance judgement for a mixture of nuclides, as also given in RS-G-1.7. On the other hand, the Standards Committee (SC) of the Atomic Energy Society of Japan (AESJ) started examining a non-governmental standard of a clearance inspection method in May 2003. A subcommittee of the SC prepared a draft report in July 2004 and the SC is planning to release the final report of the standard in 2005.

In the case of clearance inspection, a low radioactivity results in many uncertainties (errors) of measurement. On the other hand, there may be a serious uncertainty (scattering) beyond the order of the activity concentrations and nuclide spectra used in the clearance judgement. These concerns cause...
a problem in determining how the uncertainties of measurements and nuclide spectra should be handled.

To solve the problem of how to satisfy the clearance criterion while considering the uncertainties of measurements and nuclide spectra, in this paper, a new approach to establishing an appropriate safety factor for determining the uncertainties of measurements and nuclide spectra is proposed. This approach was adopted in the draft of standard of “Monitoring for Compliance with Clearance Level” prepared by the SC of the AESJ.

2. Basic concepts

2.1. Concepts of dose criterion of clearance level

In the derivation of clearance levels, the dose criterion for clearance is of the order of 10 μSv/y, which was obtained by reducing 100 μSv/y, which is equivalent to a negligible risk level (10^{-6}/y) that nobody considers in determining their action, to one-tenth while considering the overlap of practices and exposure pathways [3]. In addition, using a stochastic approach, the 97.5 percentile of the dose distribution, which is equivalent to the high endpoint of the 95% confidence interval, was confirmed to be lower than 100 μSv/y for all important exposure scenarios [3]. Thus, the clearance level of solid materials is not strictly required to be lower than 10 μSv/y. In the BSS, the dose criterion is also described as being of the order of 10 μSv or less in one year, but is not exactly 10 μSv/y. This indicates that the clearance level for solid materials may be stochastically permitted to be higher than 10 μSv/y. Therefore, for clearance level inspection, it is contrary to the basic concepts of the dose criterion for clearance to choose a safety margin for clearance judgement merely to strictly maintain 10 μSv/y.

2.2. Conservatism involved in clearance judgement

Clearance levels for every nuclide have been derived and determined on the basis of the most serious exposure pathway. To determine whether ΣC/CL is lower than 1 requires us to consider some rare cases in which the most serious scenarios with different nuclides occur simultaneously. This conservatism in the overlap of practices and exposure pathways in clearance judgement using the relation ΣC/CL<1 is considered when a dose criterion of 10 μSv/y has been set by reducing 100 μSv/y, which is equivalent to a negligible risk, to one-tenth. This implies a double consideration of the overlap of practices and exposure pathways. In RS-G-1.7, clearance judgement using the relation ΣC/CL<1 was also required for a mixture of nuclides, which means the consideration of rare cases in the exposure scenarios. For the above-mentioned reason, it can be concluded that a safety margin is included in clearance judgement using the relation ΣC/CL<1.

2.3. Basic concepts of safety margin in clearance judgement

Taking the basic concept of the dose criterion for clearance and the conservatism involved in the clearance judgement into consideration, the following concepts of a safety margin for clearance have been adopted in a draft of the Standard for Clearance Level Inspection to be released by the SC of AESJ.

a) When the 97.5 percentile of the probability distribution of ΣC/CL, due to the uncertainties in the measurement of a key nuclide, previously assessed mean concentrations and concentration ratios of the other nuclides to the key nuclide, is ten times higher than the ΣC/CL obtained using a geometric mean of C, a factor of (97.5 percentile / 10) is required as the safety margin in clearance judgement.

b) When the 97.5 percentile of the probability distribution of ΣC/CL, due to the uncertainties in the measurement of a key nuclide, previously assessed mean concentrations and concentration ratios of the other nuclides to the key nuclide, is NOT ten times higher than the ΣC/CL obtained using the geometric mean of C, no safety margin is required in clearance judgement.

It should be noted that the confirmation of whether the frequency distribution of previously assessed concentrations and concentration ratios of the other nuclides to the key nuclide is log-normal is
required to apply a practical clearance judgement. The detailed procedure of obtaining the probability distribution of $\Sigma C/CL$ is described in the following section.

3. Calculation of probability distribution of $\Sigma C/CL$

3.1. Estimation of uncertainty

The treatment of the uncertainty of measurement of a key nuclide is deeply related to the concept of detection limits. The uncertainty of measurement is generally expressed as a normal distribution. In Japan, the detection limits for radiation measurements are defined as $3\sigma$, which is three times the standard deviation of the measurement results. In the case of a monitor checked on the basis of such a detection limit, the relative error of measurement results is always less than approximately 30% since the measurement results are usually beyond the detection limit. This indicates that an uncertainty of more than approximately 30% is not required in the measurement results.

In the U.S., the concept of the detection limit is expressed by the minimum detectable amount (MDA), as reported by Currie [5]. In this case, the detection limit cannot simply be expressed by a factor of the standard deviation, but is approximately regarded as $3.29\sigma$, which is twice the value of $1.645\sigma$. This implies that an uncertainty of more than approximately 30.4% is not required in the measurement results, which is the same conclusion as that derived from the Japanese concept of the detection limit.

On the other hand, there may be extensive scattering beyond the order of mean concentration (MC method) or concentration ratios of the other nuclides to the key nuclide (NS method). The frequency distribution of the activity concentrations and nuclide ratios can be generally expressed by a log-normal distribution and determined using two parameters: the geometric mean and geometric standard deviation. To estimate the uncertainty of the mean concentrations and concentration ratios of the other nuclides to the key nuclide, one must know the geometric standard deviation that indicates the degree of scattering, after confirming whether the frequency distribution can be fitted to the log-normal one.

If the uncertainties for the measurement of the key nuclide, the mean concentrations and concentration ratios of the other nuclides to the key nuclide can be simultaneously considered in the calculation of the probability distribution of $\Sigma C/CL$, their separate safety margins would not be required. If a relative measurement error of 30% is used in the calculation of $\Sigma C/CL$, different safety margins for every measurement are not required.

3.2. Probability distribution calculation system

From the above results, for the practical use of the basic concepts of the safety margin for clearance judgement, the probability distributions of $\Sigma C/CL$ must be calculated by assuming that the uncertainty of the measurement results can be expressed as a normal distribution with a 30% relative error, which is equal to the coefficient of variation, and the uncertainties of the mean concentrations and concentration ratios of the other nuclides to the key nuclide can be treated as log-normal distributions. The calculations of the normal distributions can be theoretically treated mathematically, but the Monte Carlo calculation is the most suitable for such a calculation of the sum of the normal and log-normal distributions.

Using the Monte Carlo calculation, the probability distribution calculation system (PDCS) has been developed, which can be used to calculate the probability distribution of $\Sigma C/CL$ by taking the uncertainties of the measurement results, mean concentrations and concentration ratios into account. PDCS can determine the probability distribution of $\Sigma C/CL$ with normalized $\Sigma C/CL$ obtained by giving C a geometric mean value. In PDCS, a key nuclide must be freely selected from among target nuclides. Both geometric mean and geometric standard deviation must be given for the mean concentration and the concentration ratio.

As an example of the use of PDCS, the probability distribution of $\Sigma C/CL$ was calculated. Figure 1a) shows sample input data. The key nuclide used for measurement was $^{60}\text{Co}$ and the uncertainty was assumed to be 30%. The nuclide assessed by the NS method was $^{90}\text{Sr}$. The geometric mean and the geometric standard deviation of $^{90}\text{Sr}/^{60}\text{Co}$ ratio were 1.0 and 17.3, respectively. The concentration of $^{60}\text{Co}$ was normalized so as to satisfy the relation $\Sigma C/CL=1$, using values of activity concentration in RS-G-1.7 as CL. Figure 1b) shows output data of the probability distribution of $\Sigma C/CL$. It can be seen
that the 97.5 percentile of the probability distribution of $\Sigma C/CL$ is 25. According to the basic concepts of the safety margin for clearance judgment, a factor of $2.5$ (=25/10) as the safety margin is required in this case. That is, if there is high uncertainty in the $^{90}\text{Sr}^{60}\text{Co}$ concentration ratio, it would not be sufficient for clearance to merely confirm the relation $\Sigma C/CL<1$. The verification of $\Sigma C/CL<0.4$ (=1/2.5) would then be required for clearance judgement.

![FIG. 1a: Sample input data for PDCS](image)

$FIG. 1a$: Sample input data for PDCS

![FIG. 1b: Sample output data for PDCS](image)

$FIG. 1b$: Sample output data for PDCS

4. Conclusion

An approach to establishing the safety margin for clearance judgment has been proposed taking the concepts of the dose criterion of the clearance level and conservativeness involved in the clearance judgment into account. For practical application of the concepts to clearance judgment, a probability distribution calculation system (PDCS) has been developed, which can enable the calculation of the probability distribution of $\Sigma C/CL$ while taking the uncertainties of the measurement results of a key nuclide, the mean concentrations and concentration ratios of the other nuclides into account. PDCS is
available on the CRIEPI web site (http://criepi.denken.or.jp/en/nuclear/download/index.html) for free use.

Acknowledgement. The author wishes to thank the members of the Sub-committee of the Standards Committee of the Atomic Energy Society in Japan for their helpful suggestions and comments.

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A natural analogue of a cementitious repository: a brief overview of a study of unique sites in Jordan

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Abstract. In the framework of the Jordan Natural Analogue Programme, four phases of study have been carried out over the last 15 years. The natural cements (ca 0.5 to 2Ma old) and their associated hyperalkaline groundwater plumes (with \textit{in situ} pH values of up to 12.9, the highest ever measured for natural waters) of northern and central Jordan are ideal natural analogues of the long-term evolution of cementitious repositories for radioactive wastes. The natural cements are found across a wide area from Syria through Israel and Jordan to Saudi Arabia. However, the area of northern Jordan contains the only known site of active hyperalkaline plumes, produced by groundwater leaching of the natural cements (rather than non-relevant leachate source rocks such as the ophiolites of Oman and Cyprus) and, as such, these sites are unique. This paper presents a short overview of the work carried out to date, an introduction to ongoing work and puts the project in perspective from a SA (safety assessment) viewpoint.

1. Introduction

The natural cements and associated hyperalkaline groundwater plumes of the Maqarin and Central Jordan areas are excellent natural analogues of cement-dominated repositories and provide the best sites currently known to examine the processes associated with the long-term behaviour of such systems (see Fig. 1). The work carried out in Phases I\textsuperscript{1} to IV (see [1-5] for details) now provides a consistent picture explaining the origin of the hyperalkaline waters (with \textit{in situ} pH values of up to 12.9, the highest ever measured for natural waters), the persistence of some of the plumes and the sequence of alteration occurring (See Fig. 2) when such leachates interact with various host rock types.

The Maqarin Natural Analogue Project was initiated in 1989 with Phase I, continuing with Phase II in 1991, Phase III in 1993 and Phase IV in 2001. The Maqarin site appears to be unique in that the hyperalkaline groundwaters in the area are the product of leaching of an assemblage of natural cement minerals produced as a result of high temperature-low pressure metamorphism of marls (i.e. clay biomicrites) and limestones. In Jordan as a whole, at least three different types of hyperalkaline groundwater alteration have been identified and they appear to represent, by analogy, three different stages in the theoretical evolution of a cementitious repository for the disposal of radioactive wastes. They are:

Stage 1) early, currently active, high pH Na/KOH leachates (Western Springs, Maqarin);
Stage 2) intermediate, currently active, lower pH Ca(OH)\textsubscript{2} buffered leachates (Eastern Springs, Maqarin);
Stage 3) late, currently inactive, near-neutral pH, silica-dominated leachates (Daba and Khushaym Matruk regions in central Jordan).

Whilst Phase I and Phase II were very much site-specific and process oriented (eg studies of the source term and its interaction with the host rock; testing the applicability of available thermodynamic data to hyperalkaline conditions; predicting the extent of high pH water/rock interaction using coupled models etc), Phase III provided a more regional perspective to the geological and hydrogeochemical evolution of the entire cementitious system.

\textsuperscript{1} Work performed within the framework of the Jordan Natural Analogue Programme Phase IV, co-funded by Andra-CEA-JNC (now JAEA)-Nagra-Nirex-SKB in association with the University of Jordan.
\textsuperscript{2} Phase I was co-funded by Nagra-Nirex-Ontario Hydro (now OPG), Phase II by Nagra-Nirex-SKB and Phase III by Nagra-Nirex-SKB-UKHMIIP (now the EA/SEPA).
FIG. 1: The basis of the analogy (from [6]).

2. Ongoing studies (Phase IV)

Ongoing studies are focussed on several topics of SA concern (essentially reflecting the open questions remaining from Phases I-III) which include:

a) detailed characterisation of the source term (i.e. cement zone mineralogy) to strengthen the analogy with industrial cements, to establish the duration of alteration and to quantify dissolution fluxes;

b) rigorous evaluation of the site-scale hydro-geological and structural characteristics at Maqarin to help determine solute flux rates through the bedrock and to establish the spatial and age relationships to further develop the scenario of episodic fracture formation, sealing and reactivation which characterises the site;

c) quantification of microbial activity and its implications in high pH conditions;

d) detailed study of clay mineral stability in the presence of high pH waters as an analogy of clay-based repository host rocks and to the use of bentonite as a back-filling material;

e) iodine elemental migration/retardation in the altered rock matrix and secondary minerals.

To provide appropriate data on the five topics noted above, seven main objectives were identified and the work will be reported shortly in [5]:

(A) definition of path lengths and description of flow pathways,

(B) dating the system, including the near-field cement hydration and the longevity of C-S-H phases,

(C) rock matrix diffusion, including the retardation of sorbing (U-Th-Ra) and non-sorbing (I) elements,

(D) source term definition and cement alteration characteristics,
3. Safety assessment considerations

Examination of the natural hyperalkaline springs/seepages and their associated altered host rocks in northern and central Jordan has provided a consistent picture of some of the likely impacts of a cementitious waste repository on the host rock formation. Evidence from Phases I - IV show that:

• the conceptual model for the evolution of a hyperalkaline plume in a host rock (both fractured and non-fractured) is largely consistent with observations at the sites studied;
• hyperalkaline pore fluid conditions generated by minerals analogous to those envisaged for cements are long-lived at these sites (in excess of hundreds of thousands of years);
• secondary cement phases (including supposedly metastable amorphous phases and gels – see also [7]) produced in the host rock during leachate/rock interaction also appear to be stable for long time periods as long as they remain isolated from the groundwater;

Briefly summarised (after [6]), the model assumes that, at the cement/host rock interface, the hyper-alkaline leachates have not yet reacted with the host rock and so have a high pH and high concentrations of Na, K and Ca, reflecting the cement porewater chemistry. As the plume reacts with the host (aluminosilicate-bearing) rock, the pH falls, as do the Na, K and Ca concentrations in the groundwater, while the concentrations of Al and Si rise fractionally. Beyond the distal edge of the plume, in the, as yet, undisturbed host rock, the groundwater pH is near neutral, the Na, K and Ca concentrations are lower and the concentrations of both Al and Si are higher than in the plume waters. This pattern has consequences for the secondary mineralogy: C-S-H phases will be found in the fractures (through which the plume has migrated) in the proximal part of the plume, reflecting the fact that the leachate has not yet reacted with the host rock and is equilibrated with the C-S-H phases which make up the cement. As the leachate moves downstream and interacts with the aluminosilicates in the host rock (and the host rock groundwater and porewater), the Al concentration increases, precipitating C-A-S-H phases. At the distal edge of the plume, the leachate has reacted with an even larger volume of host rock (and the host rock groundwater and porewater) and eventually precipitates zeolites as the Al concentration in the groundwater becomes high enough and the pH low enough [4]. As these secondary phases have much larger volumes than the primary phases they replace, the matrix porosity and flow porosity slowly decrease until being effectively sealed.

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2 Briefly summarised (after [6]), the model assumes that, at the cement/host rock interface, the hyper-alkaline leachates have not yet reacted with the host rock and so have a high pH and high concentrations of Na, K and Ca, reflecting the cement porewater chemistry. As the plume reacts with the host (aluminosilicate-bearing) rock, the pH falls, as do the Na, K and Ca concentrations in the groundwater, while the concentrations of Al and Si rise fractionally. Beyond the distal edge of the plume, in the, as yet, undisturbed host rock, the groundwater pH is near neutral, the Na, K and Ca concentrations are lower and the concentrations of both Al and Si are higher than in the plume waters. This pattern has consequences for the secondary mineralogy: C-S-H phases will be found in the fractures (through which the plume has migrated) in the proximal part of the plume, reflecting the fact that the leachate has not yet reacted with the host rock and is equilibrated with the C-S-H phases which make up the cement. As the leachate moves downstream and interacts with the aluminosilicates in the host rock (and the host rock groundwater and porewater), the Al concentration increases, precipitating C-A-S-H phases. At the distal edge of the plume, the leachate has reacted with an even larger volume of host rock (and the host rock groundwater and porewater) and eventually precipitates zeolites as the Al concentration in the groundwater becomes high enough and the pH low enough [4]. As these secondary phases have much larger volumes than the primary phases they replace, the matrix porosity and flow porosity slowly decrease until being effectively sealed.
• current predictive models and databases of the solubility of elements of interest to radioactive waste disposal provide conservative estimates of solubility (i.e. solubilities are overestimated) in the hyperalkaline conditions expected in a cementitious repository;
• the amounts of colloidal material generated at the cement/host-rock interface zone will be low – but it has so far proved impossible to access the plume ‘front’ to assess the (potentially significant) colloid populations in this zone;
• sequences of minerals predicted by thermodynamic and coupled modelling are similar to those observed in the range of hyperalkaline alteration zones studied;
• the rock matrix may be accessible to diffusion of aqueous species, even during the phase of ongoing wall-rock alteration, and ongoing studies of elemental I immobilisation promise exiting new data of direct SA relevance for safety relevant $^{129}$I;
• small aperture fractures in the host rock and cement will be self-healing, larger aperture possibly so, but more evidence is required before a definitive answer can be provided;
• tectonic effects upon fracture sealing and the site hydrology need to be considered on a repository site-specific basis;
• while evidence remains sketchy at the moment, clay stability under repository relevant hyperalkaline conditions may be poor;
• microbial populations are low, but viable, under the conditions studied with the main controlling factor possibly being limited utilisation of the available nutrients.

One of the strengths of the Jordan Natural Analogue Programme has been the possibility of studying directly, at repository scale, the \textit{in situ} impact of perturbations on the original retardation qualities of the host rock. Furthermore, these site studies have been closely coupled with ongoing laboratory (e.g. [8]) and \textit{in situ} (i.e. rock laboratory) experiments (e.g. [9]). These will eventually provide greater confidence in the transferability of data and also test the limits of applicability of such data to SA models of cementitious repository systems.

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Societal optimization of radioactive waste disposal policy: 
Problem definition and solution case in Latvia

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Abstract. An increased social activity of the modern society and its modified attitude towards technical progress, environmental and health issues has caused the need to develop new approaches in the whole radioactive waste (RW) management strategy, with the aim to gain public confidence in the safety of RW management and acceptance of advanced RW disposal projects. We propose a possible model of societal optimization of RW management to be realized in an extended environment being a multitude of physical, ecological, economic, socio-cultural, psychological and other factors. The basic elements of societal optimization are considered to be social learning, risk communication and stakeholder involvement in the decision making, thus forming a knowledge-creating community and, as the result, facilitating public acceptance of RW management policy decisions. A set of actual risk perception and management issues is considered on the basis of situation in Latvia. The proposed solution are analyzed showing: 1) the divergence between scientific safety assessment of the Baldone RW repository and the public perception, 2) the possibility to offset the perceived risk by compensations, and 3) the actuality to prevent social amplification of risk by creating a knowledge-based community via educating in nuclear safety and permanent informing about environmental radiological situation. The proposed scheme and inferences could be used for further development of integral approach towards the complex problem – public acceptance of decision-making in RW management policy, via improving general attitude to nuclear activities and RW disposal solutions.

1. Introduction

Latvia has a sole national RW near-surface repository of RADON-type foreseen for LILW-SL. Its total capacity~1880 m³, in June 2005 the total amount of deposited RW was ~800 m³, total activity ~ 408 TBq. In 2006-2010 the majority of RW will come from dismantling of the shutdown Salaspils Research Reactor (~1600 m³ gross volume); in 2005 started public hearings of the design projects for 2 additional disposal vaults (each of 1200 m³) and for integral storage facility for LILW-LL (100 m³, for 50 years), mainly for long-lived spent sealed sources to be removed also from the vaults.

Permanently growing public concern about the decision making policy in RW management and aggravated social acceptance of RW disposal policy and long-term safety has led to increasing complexity of RW management problems on the whole. Highlighting that RW disposal safety is not only a very complex scientific and engineering problem, but also a profoundly difficult social and political issue” [1, 2]), we should be aware of potential threat to RW disposal safety stressed by IAEA [3] when advanced siting projects might be denied due to misunderstanding of real safety situation.

Such increased complexity of the contemporary RW management field necessitates further development approaches to solve complex problems of social communication and decision making, with the final aim to gain public confidence in the safety of RW disposal and acceptance of advanced disposal projects. Completely recognizing the importance to observe wide spectrum of interests in society, we shall stress that “effective communication and information policy are the best means of fostering a climate of confidence which will facilitate the adoption of appropriate solutions” [4]. The derived necessity in such an integral approach to solution of social problems in the RW disposal area, namely – societal optimization of RW disposal, repository siting and safety - can be considered as a basic element of optimization of the whole RW management policy.
2. Description of the optimization approach

2.1. Public awareness level – the basic parameter of optimization

The actuality of societal issues of RW disposal safety has been already widely discussed in a series of IAEA [2-6], OECD NEA [7-8] and other documents. Recognizing the importance of informing the public on issues regarding the safety of RW management expressed in [5] requiring to make information on the safety of such a facility available to members of the public, the IAEA/OECD NEA statement to implement technical solutions in the RW area “within the framework of progressive national processes that address the expectations and concerns of citizens” [9] as well as the point that democratization of the RW issues includes education all socio-technical impacts of nuclear power [6], our approach to societal optimization is based on emphasizing the role of knowledge and information in beneficial solution of the complex socio-technical problems of RW disposal safety.

Taking into account that a) the public awareness and knowledge level about nuclear energy and RW management problems is different, b) the inherent incompleteness in data on RW disposal safety, in particular, due to inherent uncertainties in the safety assessments (SA), as the basic principles in our approach to societal optimization we choose the principles which could manage with these two qualities of knowledge and information. Thus, we use such a concept of non-linear science as self-organization, and 2) the principle of requisite variety [10] requiring that for successful development of a given system in external environment its own or inherent complexity should exceed the complexity of its environment. In such a statement of the problem one should at first to specify a real content of the meanings of: i) external environment, ii) internal complexity, and iii) a given system.

2.2. The concepts of external environment, internal complexity and stakeholder community

Taking into account crucial extension of the scope for RW safety issues, in particular, marked changes in the societal environment for decision making [8], let us define the concept “external environment” as an open non-equilibrium creation including, on the basis of the concept [11] of human's three worlds, the natural environment, the social world as well as artificial environment – a set of objects, conditions and requirements emerged as the result of human and society activities. Thus, in such an extended definition the concept “external environment” will include a multitude of physical, ecological, economical, socio-cultural, psychological and other factors. Therefore, a necessary condition for successful adaptation (of the decision-making process) to crucially changing “extended” or external environment and optimization of interactions with such environment will be predominance of humans’ internal complexity over the environmental complexity.

Growing complexity of external environment markedly displaying in conditions of the decision-making process in the RW disposal area, demands to develop advanced approaches of management societal requirements for RW disposal. In this task one should especially distinguish two basic factors ways - information and knowledge - via we relate to our environment by self-organization processes [12]. If we take into account that the knowledge about the world contains, among them, such specific components as: a) knowledge about ourselves, and b) possible interactions between subjects, then for the recently actualized issue of stakeholder involvement in RW management [7,8,13,14] one can propose a way of further development of this issue, aimed as quite as possible to reveal relations between different stakeholder categories and concerns and, as the result, to find out possible forms of self-organization of various stakeholder categories into a harmonized stakeholder community having common strategic aims. Such a joint stakeholder community including all involved parts participating in decision making is considered as the given system being opposed to the external environment.

2.3. Social learning and optimization of risk perception

Thus, our task of societal optimization can be deduced to the need to elevate internal complexity of the joint stakeholder community. Viewing at the knowledge as a complexity factor, all available forms of stakeholder involvement, their education and mutual interactions can be classified as mechanisms of societal optimization, increasing the internal complexity. Firstly, it can be reached via social and mutual learning, thereby activating and diversifying interaction between stakeholders. As a key mode of this interaction can be seen the learning by operators and regulators to contact with other stakeholder groups - with the aim to elevate their knowledge level as well as to enhance mutual understanding. As
the knowledge is able to self-organize, the whole process of mutual learning and educating of stakeholders could emerge in a knowledge creating stakeholder community capable to use novel [15] communication and knowledge management forms at all levels of decision-making.

Nowadays the role of social learning rises also in the risk communication - by noting the importance of uncertainties management in confidence building of safety assessments as well as the key role of the unknown factors [16] in the determining the risk perception by public, taking into account that the basic component of social learning – adaptation – by handling uncertainty [17] - can replenish deficiency in the necessary information. Thus, as the perceived risk of repository could be regarded as a function of knowledge of repository issues [16], the role of social learning in the solving the risk perception issues displays else in such a way. Namely, in line of assumption that the unknown factor of perceived risk can be diminished via that mode of social learning where affected communities become familiar with nuclear issues, we also stress another side of social learning, namely, the ability to understand community perception of the risks, in particular, by identifying public concerns. Such ability will allow to incorporate these concerns in the decision-making mechanism, thereby raising the decision-making capacity of a stakeholder community and succeeding the public acceptance.

3. Societal problems in RW management in Latvia

3.1. Divergence between SA conclusions of the Baldone RW repository and the public perception

Recently Latvia has encountered with serious problems in management of societal issues of RW safety risk. Though the conclusions reached by the long term safety study [18] of Baldone RW near-surface repository performed by the consortium CASSIOPEE - show small risk level created by this site, the public attitude to this facility is negative or, at best, neutral. Such a perception is felt to be a consequence of the period, when many questions about the use of radioactive materials and RW disposal were classified as those creating fear in a population and leading to a negative perception of these questions both from local governments and population. As a consequence of association between the risk from radiation and the nuclear accidents one tries to implement the concept of the null risk, i.e., a demand that any undesirable event must have a risk of occurrence near zero. This leads to negative perception of any activity that could possibly have any pessimistic consequences.

3.2. Compensation to municipality: the purpose and possible mechanisms

On the basis of negative attitude to the Baldone RW repository, the municipality has demanded compensation for the fact that the RW repository is in its territory, for the consequences of fear raised in the population by the presence of the repository that is claimed to result in fewer investment and decreased of competitiveness. As new risks to society now are offset by various rewards [7, 16] (compensations, incentives, etc.), the Latvian authorities acknowledge such demand, with the aim to stimulate more positive perception of RW management activities. In particular, for the opinion poll to be conducted before public hearings for construction of new storage and disposal spaces in the repository territory, introduction of efficient compensation mechanism can serve as a positive factor.

Nature resource tax for import of radioactive materials could be one of mechanisms of implementing the principle “polluter pays”. The amount of tax is justified, in order to make a minor impact on economic development, while stimulating re-export of radioactive materials and return of disused sources to producers. For the best possible solution for this issue were proposed following 3 versions:

1) all receipts from the Nature resource tax are counted in the Baldone municipality budget,
2) annual payments from the State budget to the municipality budget, and
3) no payments, but instead to perform total risk analysis and other incentives.

As the result of analysis of these versions, the Cabinet accepted payments from the State budget.

3.3. Stakeholder involvement, social learning and educating

With the aim to prevent further social amplification of the imagined risk for the Baldone RW repository, to provide acceptance of further siting projects, to promote wide participation in decision making process and to increase the transparency of radiation safety policy on the whole, Latvia have started activities related to the stakeholder involvement [14]. In the RW management area the major stakeholders are local municipalities and non-governmental organizations. Recently Latvia has
established legal requirements (via the Law on EIA, the Regulations on the EIA procedure and the Licensing Regulations) on collection of opinions from stakeholders, thereby increasing the confidence to the state authorities. In order to create a knowledge-based community, the national Regulatory Authority - Radiation Safety Centre (RDC) – has started regular educating of teachers of physics as well as pupils in radiation safety issues, by lectures and published documents. To improve the public attitude of Baldone RW repository safety, the RDC has included in the website the early warning monitoring data including also those of the Baldone site, thereby promoting the stakeholder confidence and trust as the basic factors of efficient solution of social issues of RW disposal safety.

REFERENCES

Radioactive waste management in Mongolia

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Abstract. This paper addresses the management of radioactive wastes in Mongolia. In particular, the respective national legislative infrastructure and the organizational infrastructure are outlined. The special radioactive waste storage facility in the isotope centre are described.

1. Introduction

Radioactive waste is divided into two major categories, high-level radioactive waste and low-level radioactive waste. Low-level waste is a very broad category containing many different types of waste and a wide range of concentrations of radioactive materials. Low-level radioactive waste is generated at commercial facilities such as nuclear power plants, hospitals, and research institutions. The concentration of radioactive materials in low-level radioactive waste can vary. Precautions that must be taken when handling low-level waste depend on the radionuclides present and their concentrations. There are no nuclear power plants and research reactors in Mongolia. Radioactive wastes are generating from industry, research and medical uses. Radioactive waste management activities are carried out in the Isotope Center. Mongolia does not produce radioactive material. The amount of radioactive waste is respectively low and mostly it is generating from the spent sources of medical and industrial practices. Spent sources as a waste or disused, abandoned and orphan sources are have storing in the Isotope Center.

2. Legislative and organizational infrastructure

The Isotope Store Center (Waste Management and Transportation Service) is a radioactive waste management and transportation service unit of Mongolia, which belongs to the Nuclear Energy Commission (NEC), Government of Mongolia. The NEC and Nuclear Research Center at the National University of Mongolia has established procedures for authorization and enforcement for practices and for control of radiation source users in Mongolia. The legislation provides adequate empowerment of the Regulatory Authority and national radiation protection infrastructure. The Radiation Safety law came into enforce in July 2001. Draft regulations (Basic Radiation Safety Standards, Basic Regulation on Radiation Protection and Safety, Radioactive Waste Management Regulation, Regulation Radiation Safety for mining and milling radioactive ores) based on recommendations of BSS have been prepared. The Regulation on Safe Transport for Radioactive Material (1987) had been used. The Isotope Center of the Nuclear Energy Commission, which is located about 20 km from Ulan Bator, is responsible for the safe storage of radioactive waste and safe transport of radioactive material in Mongolia. The Isotope Center of the Nuclear Energy Commission had been established in 1987.

3. Storage facility in the Isotope Center

The construction of the storage facility of the Isotope Center (Fig.1) is of reinforced concrete and is framed with walls and a flat roof also made of concrete, suited for the warm and also cold climate. The Isotope Center has two storage facilities. The 1st storage facility (Fig.2) has an area of 24m x 9m, divided internally in a number of rooms and areas for different purposes, and has 12 storage wells for spent sealed sources. Also, this storage facility has a heating system, sewage system and lifting system. The capacity of this well is 1.2m x 1m x 1m, and concrete walls are 20 cm thick, the concrete base is 30 cm and the concrete cap is 20 cm thick. The 2nd storage facility has an area of 11m x 6m, which has 6 wells. The storage facility is designed for storage of respectively high activity and long-lived radioactive solid sealed sources such as $^{60}$Co, $^{137}$Cs, Pu-Be, $^{226}$Ra etc. The well size is 2.20m x
1.80m x 1.80m, and concrete walls are 30 cm thick, the concrete base is 13 cm thick and the concrete cap is 20 cm thick.

**FIG. 1: Radioactive waste disposal**

A special cemetery of soil contaminated by $^{90}$Sr is located at the isotope center. In 1993 an area contaminated by $^{90}$Sr was found in Ulan Bator city and the soil was removed and decontaminated. This incident and the decontamination was reported to the IAEA.

**FIG. 2: Storage facility**

The soil waste was placed in industrial drums. The drums had capacities of 200l and had corrosion resistant internal surfaces coated with bitumen. The waste required in total 93 drums and, after sealing, was placed in the waste storage site. For this reason, special 6 storage cabins (each cabin is 1.73m long, 1.58 m wide and 2.78 high) made of concrete walls were built at the storage site at the Isotope Center. The concrete wall was 10 cm thick and the concrete base and cover were 20 cm thick. To protect it from water effects, it was covered with plastic sheets. This cemetery is located within the fence of the Isotope Center near the 2nd storage facility. Recently, in Article 3.1.8 of the Law of Mongolia on Radiation Protection and Safety prescribed "Radioactive waste" as following: "Radioactive waste" means material, in whatever physical form and for no further use that contains or is contaminated with radioactive substances and has higher a activity concentration than a level specified in the Standards. The management and disposal of radioactive waste are specifically dealt with under articles 5.1; 5.6; 5.7; 6.1.3; 8.1.1; 8.1.2; 15.1.1; 15.1.2; 20.1.1; 20.1.2; 20.2 of the Law of Mongolia on Radiation Protection and Safety:
(a) the Nuclear Energy Commission (Commission), under the auspices of the Member of the Government charged with science issues, shall be responsible for development of policy and regulatory control for the activities relating to development of nuclear research and technology, use of radiation sources, and to ensure radiation protection and safety (5.1.);

(b) the Commission shall have a special waste storage facility, responsible for safe storage, transportation and unused radiation sources and radioactive waste within the country (5.6.);

(c) the special facility shall be an object of the state protection (5.7.);

(d) the NEC shall exercise the power: to charge the activities of mobilizing, depositing, transporting and reposing unused radiation sources and radioactive waste within the country (6.1.3);

(e) the licenses shall be granted, to a person, business entities, and organizations that satisfy the requirements stipulated in this law and other relevant legislation, for the following activities relating to radiation: to order, import, export, receive, distribute, construct, allocate, use, keep in possession or use, keep in repairing or transferring period, sale, rent, produce, process, reprocess, exploring, acquire, compose project for use, discard, deplete, transport, store, deactivate (decontaminate), and to bury radiation sources (8.1.1.);

(f) to explore, mining, process, enrich, import, export, transport of radioactive ore, and to bury waste, and land rehabilitating activities after mining of radioactive ore (8.1.2.);

(g) citizens, business entities, organizations shall comply with the following requirements to store radiation sources:

15.1.1. to store, in compliance with respective rules and regulation for radiation protection and safety, in special waste storage site recommended by the Commission taking into consideration radiation sources’ specific characteristics;

15.1.2. to transfer radiation sources, which do not meet technological or safety requirements, or unused sources, to the state centralized waste storage facility stipulated in the article 5.6 of this law;

(h) citizens, business entities, organizations shall comply with the following requirements for reposing radioactive waste: To have conclusion of the state radiation inspector and permit from the Commission and other relevant state inspectors of respective profession. (20.1.1.);

(i) to repose radiation waste under supervision of the Commission and other relevant state inspectors of respective profession. (20.1.2.).

Table 1: Results of measurement of total alpha and beta activity in Ulan Bator city soil

<table>
<thead>
<tr>
<th>No.</th>
<th>Assembled places of soil samples</th>
<th>Total beta activity, Bq/ kg</th>
<th>Total alpha activity, Bq/ kg</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Naran</td>
<td>318.2±11.4</td>
<td>55.5±1.5</td>
</tr>
<tr>
<td>2</td>
<td>Ulaankhuarin</td>
<td>318.2±11.4</td>
<td>3.7±0.7</td>
</tr>
<tr>
<td>3</td>
<td>Nepti baaz</td>
<td>229.4±3.7</td>
<td>-</td>
</tr>
<tr>
<td>4</td>
<td>Bayanzurkh</td>
<td>281.2±7.4</td>
<td>7.4±0.7</td>
</tr>
<tr>
<td>5</td>
<td>Amgalan</td>
<td>370.0±11.1</td>
<td>25.9±0.1</td>
</tr>
<tr>
<td>6</td>
<td>Khurkhree</td>
<td>318.2±11.1</td>
<td>7.4±0.7</td>
</tr>
<tr>
<td>7</td>
<td>Uildver combinat</td>
<td>292.3±11.1</td>
<td>-</td>
</tr>
<tr>
<td>8</td>
<td>Tasgani ovoo</td>
<td>310.8±11.1</td>
<td>33.3±0.1</td>
</tr>
<tr>
<td>9</td>
<td>Sharkhad</td>
<td>336.7±11.1</td>
<td>22.2±0.1</td>
</tr>
<tr>
<td>10</td>
<td>Yarmag</td>
<td>377.4±11.1</td>
<td>11.1±0.1</td>
</tr>
<tr>
<td>11</td>
<td>Tsagaankhuarin</td>
<td>377.4±11.1</td>
<td>18.5±0.1</td>
</tr>
<tr>
<td>12</td>
<td>Tolgoitiin etses</td>
<td>236.8±7.4</td>
<td>-</td>
</tr>
<tr>
<td>13</td>
<td>500 ail</td>
<td>329.1±11.1</td>
<td>33.3±0.1</td>
</tr>
<tr>
<td>14</td>
<td>Zaisan</td>
<td>314.5±11.1</td>
<td>48.1±0.2</td>
</tr>
<tr>
<td>15</td>
<td>Denjiin 1000</td>
<td>240.5±11.3</td>
<td>29.6±0.4</td>
</tr>
<tr>
<td>16</td>
<td>Hailaast</td>
<td>181.3±3.7</td>
<td>11.1±0.7</td>
</tr>
<tr>
<td>17</td>
<td>Chuluuntseg</td>
<td>222.0±3.7</td>
<td>11.1±0.7</td>
</tr>
<tr>
<td>18</td>
<td>Dari ehiin ovoo</td>
<td>210.9±3.7</td>
<td>22.2±0.1</td>
</tr>
<tr>
<td>19</td>
<td>12-th khoroolol</td>
<td>144.3±3.7</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>Average</td>
<td>284.7±8.8</td>
<td>23.7±0.4</td>
</tr>
</tbody>
</table>
In the selection of samples for the analyses of soil, relatively equal platforms were chosen: equal places by the size of 1 m² at a depth of 5 cm from surface soil. Table 1 lists the name of the districts where soil samples were taken.

Mongolia has a large perspective of the application of nuclear methods for the solution of a variety of urgent matters, thus strengthening the national economy.

REFERENCES

A procedure for safety verification of low-level radioactive wastes

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Abstract. Homogeneous and uniform solidified wastes (HUSW) and container-filled and solidified wastes (CFSW) out of low-level radioactive wastes (LLW) produced from the operation of nuclear power plants (NPPs) have been already buried and disposed of at the Low-Level Radioactive Waste Disposal Center of Japan Nuclear Fuel Limited (JNFL) in the Rokkasho-mura. Since Japan Nuclear Energy Safety Organization (JNES) was established in October 2003, JNES has carried out the jobs of safety verification for these waste packages and preparing procedures for waste safety verification. This paper reviews the conventional procedures for waste safety verification and explains the requirements to be prepared for continuous use of the scaling factor method or average radioactivity concentration method (hereinafter called “SF”) for new CFSW that are required after the establishment of JNES. We have placed our emphasis on the transparency of these jobs and published all results of examination to the public.

1. Introduction

Wastes solidified in drums each of 200 litres out of LLW produced in the operation of NPPs are buried and disposed of at the Low-Level Radioactive Waste Disposal Center of JNFL in Rokkasho-mura. These waste packages are classified into HUSW and CFSW depending on different wastes contained in the drums and solidification methods, as shown in Fig. 1.

<table>
<thead>
<tr>
<th></th>
<th>HUSW</th>
<th>CFSW</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sketch</td>
<td><img src="image" alt="Sketch of HUSW" /></td>
<td><img src="image" alt="Sketch of CFSW" /></td>
</tr>
<tr>
<td>Waste concerned</td>
<td>Liquid wastes (concentrated waste, spent resin, etc.)</td>
<td>Solid wastes (metals, concrete, etc.)</td>
</tr>
<tr>
<td>Safety Verification Procedure for Continuous use of SF</td>
<td>Prepared for wastes after 1991</td>
<td>Not prepared</td>
</tr>
</tbody>
</table>

FIG. 1: The outline of low-level radioactive waste packages

The safety requirements that these waste packages must meet are stipulated in the Law [1] and the related ministerial ordinances. The safety verification of LLW packages, which objective is to confirm that each package meets the prescribed requirements, is carried out before carrying into the disposal center of JNFL. To verify an object as a radioactive waste, it is necessary to assess the concentration of radioactivity of the specified nuclides in individual waste packages, but as a result of deliberations in the Nuclear Safety Commission (NSC), it has been decided to use the SF as a method for determining the concentration of radioactivity of nuclides whose radioactivity is hard to be directly measured. The current SFs for HUSW and CFSW were set up based on measurement and analysis of LLW discharged from NPPs operations before fiscal years 1990 and 1997, respectively. Therefore, this method must be used by properly verifying if it can be continuously used in determining waste packages produced after...
the setting up. Furthermore, for a waste package produced by a new manufacturing technology, it must be verified if the waste package meets the technical requirements, and at the same time a concrete procedure for the verification must be properly prepared.

Since the establishment of JNES in October 2003, the job of verifying wastes for the disposal of LLW has been assigned to JNES by law. This paper outlines the job of waste safety verification conducted by JNES and summarizes the actual conditions of examining the continuous use of the SF.

2. **Technical requirements and conventional procedures for waste safety verification**

The technical requirements that waste packages must meet are stipulated in the Rule [2] and Bulletin [3] and summarized in Fig. 2 for CFSW.

- **Solidification material**
  - Cement: cement specified in JIS R 5210 (1992) or JIS R 5211 (1992), or cement having the quality equivalent to or better than the above.

- **Harmful vacant space**
  - No harmful vacant space shall remain.

- **Container**
  - Container specified in JIS Z 1600 (1993) or container having the strength and sealing property equivalent to or higher than the above one.

- **Surface contamination density**
  - Radioactive material that releases alpha rays: 4Bq/cm² or less
  - Radioactive material that does not release alpha rays: 4Bq/cm² or less

- **Mixing of solidification material**
  - Solidification material shall be sufficiently mixed in a uniform state and without uneven distribution

- **Solidification material**
  - Solidification material shall be sufficiently mixed in a uniform state and without uneven distribution

- **Fill wastes so as to produce a uniform state**
  - Fill solidification material so as to be unified into radioactive waste in container.

- **Radioactivity concentration**
  - Radioactivity concentration shall not exceed the maximum concentration described in an application for a license

- **Surface dose equivalent rate**
  - Surface dose equivalent rate shall not exceed 10 mSv/h.

- **Significant damage**
  - There shall not be significant damage.

- **Marking and reference number**
  - Marking for indicating a radioactive waste
  - Marking for indicating a surface dose equivalent rate (in color zone)
  - Reference number

- **Period of 6 months or more**
  - RW contained in a waste package shall pass 6 months or more from the production of the waste package.

- **Burial load resistance**
  - Solidification material shall have the strength enough to withstand a load which may be produced in burial.

#### FIG. 2: Technical requirements for container-filled and solidified wastes (CFSW)

The safety verification is carried out by the inspection institute in consideration of the manufacturing methods of the packages provided by the operator. The regulating authorities notified the then Nuclear Safety Technology Center (NSTC) about the procedure that stipulates the details of the verification, because NTSC used to be an appointed institute for the verification.[4], [5]. Since the establishment of JNES, the verification job has been transferred to JNES by law. The scheme of safety verification is shown in Fig. 3.

3. **Flow of job for verifying waste packages**

#### FIG. 3: The safety verification procedure for LLW by JNES
At the present time, JNES not only has carried out waste safety verification as an inspection institute, but also has been in charge of evaluating the competence of technical requirement for new waste packages and examining the continuous use of the SF. The flow of this job for new waste safety verification is explained below.

First of all, an operator requests JNES to evaluate the adequacy of a proposal for continuous use of the SF or a method for verifying a waste package as a new waste one, and then JNES internally starts examining the request. In the process of examination, JNES asks men of learning and experience for their opinions in an advisory committee set up in JNES for technology of waste safety verification, and results of investigation are disclosed to the operator and at the same time published to the public. In addition, JNES reports to the regulating authority about the results of investigation. The operator shall apply for actual waste safety verification in accordance with the procedures whose adequacy has been verified by JNES. JNES inspects the waste safety verification and issues a certificate of safety verification when the waste package meets the technical requirements.

4. Method for determining radioactive concentration

The method for determining a radioactive concentration is an important matter for waste verification especially from a viewpoint of safety. The NSC investigated methods for determining a radioactive concentration of the specified nuclide in waste packages, and as a result, it has been decided to use the SF as a method for a nuclide whose radioactivity is hard to be directly measured. This method consists of collecting representative samples from among waste packages or LLW produced in the past and setting up relative value to key nuclide mainly from results of radiochemical analysis of these samples. When it was set up, the SF was permitted to apply for waste packages produced until fiscal 1990 for HUSW and for those until fiscal 1997 for CFSW, respectively. Then, the regulating authority decided a way for continuous use of the SF for HUSW produced after fiscal 1991. This time, JNES has established a way for continuous use of the SF for CFSW produced after fiscal 1998.

4.1. Results of continuous use of SF for HUSW

The basic idea on continuous use of the SF given below was approved in the report of April 2, 1992 prepared by the NSC:

“It is thought that the SF established does not change from the first, with the exception of (1) large-scale change of reactor structural materials, (2) fuel damage occurrence, and (3) change of solidification technology process. Therefore, the existing SF can be used, if a change in the SF is verified by continuing the radiochemical analysis of representative samples, and no significant difference is found between analysis results and existing values. In addition, different waste packages to be produced in the future may be determined for their radioactivity concentrations directly by radiochemical analysis method of original concentrated liquid waste prior to solidification.”

In accordance with this idea, if a sample per year is taken from the liquid tank before solidification treatment and a nuclide ratio obtained from the radiochemical analysis does not exceed 10 times the existing SF value, it is judged that the SF can be continuously used.

4.2. Establishment of judgement on continuous use of SF for CFSW

The basic idea on continuous use of the SF for CFSW to be produced after fiscal 1998 is in principle the same as that for HUSW. Further, it is expected that any change of the SF in a solid waste can be verified with a change in a liquid waste (concentrated liquid waste). This idea is based on the following fundamental ones:

A: grasping the change of nuclide ratios in concentrated liquid wastes enables us to understand the change of nuclide ratios in reactor water;
B: grasping the change of nuclide ratios in reactor water enables us to understand the change of nuclide ratios in solid wastes;
C: therefore, grasping the change of nuclide ratios in concentrated liquid wastes enables us to understand the change of nuclide ratios in solid wastes.
In accordance with this idea, the adequacy of predicting the SF change for CFSW with a concentrated liquid waste is evaluated and examined for every nuclide.

4.2.1. Fission product nuclides

As fuel damage hardly occurs in recent plants, the concentration of I-129 in concentrated liquid wastes is extremely low, and therefore it is very difficult to detect I-129 by radiochemical analysis. In this event, it is judged that the SF can be continuously used if the following conditions are met:

(1) when there is no fuel damage, and

(2) even in the case of fuel damage, when it can be confirmed from the behaviour of I-131 concentration in reactor water that the influence of fuel damage does not reach a level of damage to produce the significant change of the SF.

The SF of FP nuclides including Sr-90 and all α nuclides will be able to judge in the similar method.

4.2.2. Radioactive nuclides activated by radiation

Ni-63 and Nb-94 including Tc-99 have good correlation with Co-60. SF of these nuclides obtained by the radiochemical analysis of concentrated liquid wastes has shown a good agreement with one of solidified wastes. It is adequate to judge if the SF can be continuously used, from the behaviour of Ni-63, Nb-94 and Tc-99 with Co-60 as a key nuclide in liquid waste.

4.2.3. H-3 and C-14

The H-3 concentration in reactor coolants is almost constant through individual operation cycles, and there is no change of H-3 concentration with time in waste packages. The average H-3 concentration in CFSW has a great safety margin for maximum allowable concentration. And since the main process for producing C-14 in BWR is radio activating oxygen in reactor coolants, it is judged that there is no great variation factor in a production quantity of C-14.

Based upon the above examination, the existing SF can be continuously used by checking the influence of fuel damage if newly occurred.

5. Conclusion

Since it was established in October 2003, JNES, as an inspection institute, not only has conducted waste verification, but also has been in charge of evaluating the conformity of new waste packages to the technical requirements and examining the continuous use of the SF. With its establishment as a turning point, JNES has rearranged the conventional methods for waste verification, and prepared the conditions for continuously using the SF for new container-filled solids produced after 1998 and the procedures for verifying the waste packages produced by new manufacturing methods. These results are disclosed to the business operators and at the same time published to the public. JNES will work keeping the business transparent for new waste packages to be produced in the future.

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[5] Notification for implementing verification of CFSW (Notification No. 43 on waste regulations of September 27, 1999 by Nuclear Safety Bureau, the Science and Technology Agency).
Developing and integrating the technical basis supporting the HLW geological disposal programme in Japan

H. Ishikawa, H. Umeki, K. Miyahara

Japan Nuclear Cycle Development Institute (JNC), Ibaraki

Japan

Abstract. The implementation plan for the Japanese HLW disposal programme is being developed with particular emphasis on ensuring safety at all stages, through construction, operation and closure, by appropriate selection of a site and an associated repository design. This should increase confidence in the technical foundation for both implementation and regulation. The safety case at all stages would thus be based on sound science and engineering, which would be continuously strengthened by focused research and development (R&D). This paper overviews the current R&D activities of JNC, including the progress of two generic underground research laboratory projects, in which the surface-based investigation stage is almost complete.

1. Introduction

In 1999, the Japan Nuclear Cycle Development Institute (JNC) published a technical report (“H12”) that compiled all relevant research and development on HLW disposal and outlined the feasibility of such a geological disposal program in Japan [1]. Based largely upon the technical achievements of H12, the implementation phase for HLW disposal was initiated in the year 2000; the law regulating implementation (“the Final Disposal Act”) was passed and the implementing entity, NUMO (Nuclear Waste Management Organization of Japan), was established.

NUMO has defined a stepwise site selection procedure following the requirements of the Final Disposal Act and has documented approaches to characterizing and comparing sites [2] and to developing repository concepts tailored to given siting environments [3]. The discussions on establishing safety regulations have been also initiated by the Nuclear Safety Commission (NSC) and the Nuclear and Industrial Safety Agency (NISA).

In accordance with the new framework specified by the Atomic Energy Commission of Japan in 2000 [4], JNC continues to be responsible for R&D activities aimed at enhancing the reliability of disposal technologies and safety assessment methodologies and associated databases. JNC has thus actively promoted technical R&D with a view to contributing to both the implementation of disposal and the formulation of the safety regulations (see Fig.1).

FIG. 1: Contribution of R&D to Implementation and Safety Regulations

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2. Post-H12 Research and Development

2.1. URL projects

One of JNC’s key roles is to establish and demonstrate site characterization methodologies based on investigations in two purpose-built generic URL (underground research laboratory) projects: one at Mizunami (MIU) in crystalline rock and the other at Horonobe in sedimentary rock. These URLs are research facilities and are distinct from the site-specific underground test facilities planned to be constructed at potential waste disposal sites by NUMO.

The URLs provide a wide range of possibilities for underground research by universities or other research institutes, as well as serving as a tool for enhancing public understanding of R&D activities related to geological disposal.

The output obtained from the URLs will be widely published and is expected to make a timely contribution as follows to the disposal programme and to the establishment of safety regulations:

i. Techniques will be developed for characterizing the geological environment from the surface to deep underground, based on investigations from the surface (phase 1). This will take into account requirements relating to the design of the disposal system and safety assessment.

ii. Data obtained from investigations during the excavation phase (phase 2) will serve to verify the results from the surface-based investigation phase and characterize the evolution of the geological environment during drift excavation. Models will also be refined based on the geological data.

iii. Detailed investigations in the underground facility (phase 3) will contribute to refining geological investigation techniques that take into account the requirements of disposal system design and safety assessment. Data will also be compiled on the geological conditions in order to verify the reliability of models.

The MIU project commenced in 1996. The surface-based investigations at the MIU construction site located on land owned by Mizunami City began in March 2002. Construction of the Mizunami URL (phase 2) began in July 2002. According to the current design, the Mizunami URL will consist of two 1,000 m deep shafts. Shaft excavation commenced in July 2003 and, as of October 2004, entrance structures to the shafts had been constructed and shafts had been excavated to a depth of 50 m. Construction of the Mizunami URL is expected to be completed around 2010.

At the Horonobe URL, surface-based investigations (phase 1) have been ongoing since 2001. Following regional investigations, an area for intensive site investigations (URL area) was identified. The land for construction of the facility (URL site) was then acquired within the URL area and site preparation began in July 2003. According to the present conceptual design, the Horonobe URL will consist of two 500 m deep access shafts. Shaft excavation (phase 2) will start in 2005. Construction of the Horonobe URL is expected to be completed around 2010.

2.2. Developments in repository engineering and performance assessment

R&D on repository concepts and performance assessment has been focused on key issues and uncertainties identified in H12, which will be important for developing a convincing safety case. Basic studies and experiments in two surface-based laboratories, ENTRY for mainly engineering-scale experiments and the QUALITY facility for experiments using radionuclides, have been carried out to improve understanding of the long-term behavior of the geological disposal system and to complement results obtained from the two URLs.

In order to enhance confidence in the H12 design concept, the robustness of key safety functions of the engineered barrier system (EBS), such as physical containment by the overpack and retention of radionuclides in the bentonite buffer, have been studied for saline groundwater expected at coastal sites and for the high-pH plume which might be caused by use of cementitious material in the repository. Work has also been initiated to improve understanding of coupled THMC processes in the near-field, in order to better evaluate system evolution. This will provide a technical basis for possible pre-closure monitoring aimed at confirming the initial conditions for the post-closure safety assessment.
Assessment of the potential impact of natural perturbation phenomena (volcanic activity, fault movement, uplift/erosion and climate/sea-level change) plays an important role in increasing confidence in the safety of the disposal system. In this regard, the simplified and stylized approach in H12 to assessing such scenarios has been improved by developing more realistic models that can simulate safety-relevant impacts of these perturbations. This approach can also provide useful guidance to help focus site investigation programs.

Studies on model uncertainty are focused on processes related to key safety functions, such as:

- long-term glass dissolution;
- diffusivities and porosities of gouge within fractures;
- colloid-facilitated radionuclide transport in fractured rock;
- complexation of key radionuclides with natural organic substances.

Development and updating of key databases includes determination of thermodynamic and sorption data for safety-relevant elements under relevant conditions - including hyperalkaline and saline systems. In order to ensure quality, standardized data acquisition methods for these databases are under development.

3. Integration of technical achievements into a knowledge base

Through the surface-based investigations in the Mizunami and Horonobe projects (phase 1), integration of work from different disciplines into a “geosynthesis” has been illustrated and is planned to be developed further in the underground facilities at these sites (phases 2 and 3). These projects also serve for developing and testing the tools and methodologies required for site characterization. Further know-how will be gained through participation in foreign underground laboratory projects, transfer of experience from these projects to Japan and tailoring it to Japanese conditions and requirements. This experience represents an important knowledge base, which is obviously important for the implementer but is also needed by the regulator, in order to assess how key site characteristics are derived and what uncertainties are associated with this process.

Such site investigations and subsequent repository design studies provide input for associated safety assessment. In turn, safety assessment has an important role to play in guiding site investigations and repository design activities by quantifying key sensitivities and uncertainties. An integrated program in which these topics are developed iteratively is thus needed. Such integration requires experienced multidisciplinary project teams who should also have a wide perspective provided by participation in multinational projects and familiarity with other national HLW programmes.

JNC has developed the JNC Geological Disposal Technical Information Integration System (JGIS), to facilitate program integration and sharing of technical information between the site investigation, repository design and safety assessment teams. JGIS is an archive system in which a relational database stores technical information in the form of a structured flowchart that systematically represents the structure of research activities. In addition to the development of such a structured knowledge base, JNC will publish a state-of-the-art report, “H17”, in autumn 2005, which will document R&D results based on the phase 1 activities in the two URLs and studies in ENTRY and QUALITY, as well as future development of a knowledge management system for geological disposal.

4. Conclusions

JNC has been promoting R&D aimed at increasing confidence in the technical basis provided in H12 by making maximum use of its infrastructure. The experience gained in the JNC URL projects at Mizunami and Horonobe will contribute to the NUMO site characterization program by providing tools, experience and manpower as and when required. In order to develop an optimized design and safety assessment basis for specific sites, variants of the “requirements management system” being investigated by NUMO, which could be integrated with the development of “knowledge management” and “quality management” systems, are being considered. Through development of a quality-assured knowledge base, JNC as an independent third party can also act as a valuable resource for both the implementer and the regulator. JNC will be integrated with the Japan Atomic Energy Research
Institute to form a new organization in October 2005. A structured approach to managing technical knowledge on geological disposal is also critical in this regard, allowing the new organization to preserve a valuable legacy of intellectual property and to continue to play a central R&D role in the HLW disposal program.

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New regulatory requirements for radioactive waste management

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Abstract. In the late 90s the construction of basic radioactive waste management facilities were concluded in the Slovak Republic. The Bohunice treatment and conditioning centre was commissioned and began the operation at 2001 and in the same year the operational license for Mochovce near surface disposal facility was granted by UJD SR. The licensing process and the provisions for construction and operation of radioactive waste management facilities including repositories were set down in the Atomic act, which entered into force in 1998. The Slovak Republic joined the European Union on 1st May 2004 as one of 10 “new” countries. It was necessary to transpose the European legislation into Slovak legislative documents, also in the area of nuclear energy. The new Atomic act which entered into force on 01.12.2004 and respective regulations set down the provisions for licensing process and for whole lifetime of radioactive waste management facilities. There are many new provisions in the new Atomic act and in the new set of respective regulations, which are elaborated on the basis of feedback gained from operation of Mochovce near surface disposal facility. The paper presents the development of new legislation for radioactive waste management with the emphasis on disposal of radioactive waste in the connection with the experience gained with the construction and operation of these facilities.

1. Introduction

The legal basis for peaceful use of nuclear energy and namely the part concerning the radioactive waste management and decommissioning is developing considerably during the last decade in the Slovak Republic. This is caused by the fact, that the radioactive waste strategy has changed and the disposal option for low level and intermediate level radioactive waste was adopted. Original soviet design concept for the radioactive waste management supposed the collection, pre-treatment and interim storage of all radioactive waste produced at the site during whole operational period. Such concept postponed the conditioning and disposal of operational waste to the decommissioning stage.

The government approved the new strategy for radioactive waste management in Slovakia and after its update it can be characterized as follows:

- Effective use of current installed and verified facilities and technologies for treatment and conditioning,
- Use of cementation and bituminisation as basic technologies for liquid and wet radioactive waste,
- Use of supercompaction and incineration as basic technologies for solid radioactive waste,
- The conditioned radioactive waste produced during the operation and decommissioning of nuclear power plants that meet the acceptance criteria shall be disposed of in the Mochovce near surface disposal facility,
- The radioactive waste which is not acceptable for the Mochovce repository shall be stored at the power plants, an integral storage shall be built to allow storing of radioactive waste,
- The strategy for management of radioactive waste which does not meet the criteria for disposal in near surface disposal facility shall be elaborated, one option shall be the development of deep geological repository,
- The costs of radioactive waste management produced during the decommissioning of nuclear facilities shall be covered from the resources of State fund; the costs of radioactive waste management produced during the operation of nuclear facilities shall be covered by the operational costs.
2. Legal basis for radioactive waste management including disposal

There are two principal regulatory bodies in the Slovak republic that are responsible for nuclear safety and radiation protection. The distinction between the responsibilities of these regulators in the area of radioactive waste management is clearly stipulated in the Atomic act.

The Nuclear regulatory authority is responsible for all steps of management with the radioactive waste from nuclear installations and for conditioning and disposal of institutional radioactive waste (from Medicine, Research and Industry) and the regulatory body under Ministry of Health (State Health institute) is responsible for management of institutional radioactive waste except its conditioning and disposal.

The new requirements in the area of radioactive waste disposal were included into new Atomic act that entered into force in December 2004 and into the set of regulations that are in the final stage of development and it is supposed that they will enter into force within the end of this year.

2.1. New atomic act

This act defines that disposal of radioactive waste or spent fuel shall mean emplacement of radioactive waste or spent fuel into radioactive waste or spent fuel repository and radioactive waste or spent fuel repository is defined as nuclear installation that allows radioactive waste or spent fuel isolation, control and protection of the environment. This definition allows the retrievability of radioactive waste in all phases of its lifetime although it is not intended.

The act stipulates further the new organization responsible for radioactive waste disposal and states that disposal of radioactive waste or spent fuel may be only undertaken, based on authorization issued by the regulatory body, by a legal person independent of the originator of radioactive waste, founded or established by Ministry of Economy. The organization that carried out the disposal of low and intermediate level radioactive waste in Mochovce repository is not so far independent of radioactive waste originator and the experience gained by our regulatory body has shown that the control of radioactive waste generator by not independent organization cannot be done effectively.

In the part of act dealing with the radioactive waste management there are provisions, which are new or there are verified as good practice, the main of them are:

- The responsibility for the management of radioactive waste prior to their acceptance at the repository shall be with the originator of the radioactive waste,
- Radioactive waste must be managed so as to maintain sub-criticality, secure removal of residual heat, minimize effects of ionising radiation on operators, population and the environment and to take into account properties influencing nuclear safety such as toxicity, flammability, explosiveness and other hazardous characteristics,
- The responsibility for the disposal of radioactive waste form nuclear installations including repository closure and institutional control shall be with the state under conditions laid down by the act and other generally binding regulations,
- Repository of radioactive waste may only be located on land owned by state,
- Costs associated with the management of radioactive waste including costs linked with institutional control after repository closure shall be borne by the originator of radioactive waste,
- The regulatory body shall designate another holder of authorization of radioactive waste management in the case that radioactive waste originator is not known or it is unable to safely manage radioactive waste. The regulatory body shall define the scope of management with such kind of radioactive waste,
- Costs associated with management of radioactive waste which originator is unknown shall be born by State fund of decommissioning. If the originator is identified subsequently he shall be liable to reimburse the State fund.
The new act stipulates further the provisions for licensing process for nuclear installations. The licensing process for radioactive waste management installations has following principal steps:

Siting, construction, commissioning and operation, decommissioning stages (in case of disposal facility repository closure).

The license for radioactive waste installation siting is granted by regional civil office after the environmental impact assessment process is concluded. The licenses for further steps of radioactive waste management installations lifetime is granted by Nuclear Regulatory Authority (NRA). The safety documentation shall be prepared by applicant and it is subject of the regulatory bodies review, for nuclear safety is responsible NRA, for radiation protection Ministry of Health, for fire protection Ministry of Interior and for general safety Ministry of Labour, Social Policy and Family.

The EIA process is required before siting and decommissioning (closure) of installations for radioactive waste management.

2.2. New set of regulations

Details of the requirements on management of radioactive waste are laid down in the regulations that are also generally binding legal documents.

The regulations that are relevant for the area of radioactive waste management are under final stage of development and there are prepared as follows:

- Regulation on radioactive waste and spent fuel management, which are most relevant for this area. The regulation stipulates provisions for all steps of radioactive waste management including provisions for radioactive waste management installations. The main provisions for radioactive waste disposal includes the scope and content of safety analysis, the sensitivity and uncertainty analysis, the institutional control time period (300 years). The repository monitoring, repository closure.

- Regulation on radioactive waste shipment stipulates provisions according to international requirements and European union directives and includes provisions for import and export of radioactive waste.

- Regulation on licensing documentation contains details for preparation of documentation required for licensing of nuclear installations siting, construction, commissioning and operation, decommissioning steps and in the case of repository for its closure.

- Regulation on nuclear installations safety requirements in the process of their siting, design, construction, commissioning, operation, decommissioning and repository closure contains the set of requirements that shall be fulfilled by the applicant in the period of different stages of nuclear installations lifetime.

Some provisions for management of radioactive waste are defined in other prepared regulations for example in the regulation on quality management and in the regulation on periodical assessment of nuclear safety.

2.3. Acts linked with radioactive waste management

The act on environmental impact assessment defines the provisions for evaluation of proposals for all new nuclear facilities and also proposals for their decommissioning. The environmental impact assessment process includes hearing of citizens in local and neighbouring municipalities and in relevant cases statement of neighbouring countries.

The act on protection of public health stipulates requirements of radiation protection and all radiological limits for workers and public.

The act on State fund for decommissioning, spent fuel and radioactive waste management gives details on creation and use of fund. The fund can be used for decommissioning, management of spent fuel and radioactive waste from facilities under decommissioning for management of orphan sources and for research and development in the field of radioactive waste management.
3. Conclusions

Disposal of radioactive waste is the specific step of radioactive waste management that requires the ensuring of nuclear safety provisions for very long time. This fact requires also the special approach in preparation of legal instruments regulating the radioactive waste disposal process. This approach was used also during the preparation of the new Atomic act and the set of respective regulations. The provisions regulating the radioactive waste disposal are in most cases specific for this radioactive waste management step.

The new legal basis for disposal of radioactive waste covers all lifetime of disposal facilities from their siting to repository closure.
Consequences of a potential igneous disruption of a high-level nuclear waste repository at Yucca Mountain, USA

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Abstract. This paper identifies and evaluates a number of processes associated with the performance of man-made and natural components during a potential igneous event at the potential nuclear repository site at Yucca Mountain, Nevada, USA where existing previous DOE and NRC analyses have made highly conservative assumptions regarding the performance of man-made and natural components during a potential igneous event at Yucca Mountain, Nevada, USA. In particular, detailed thermo-mechanical impact and heating analyses show that waste packages will survive intact during the eruption phase, hence, zero release of radionuclides to the atmosphere are expected. Additional sensitivity analyses using realistic models of responses of engineered and natural barriers to the extrusive event have been carried out that demonstrate multiple factors that assure safe isolation of nuclear waste. Based on these analyses, safety assessment calculations confirm that postulated releases, even for multiple, highly speculative “what if?” conditions, are many orders of magnitude below the regulatory limits. This approach shows the benefit of the use of multiple lines of reasoning combined with best-estimate analyses in providing confidence in repository performance.

1. General approach

The probability of a future monogenetic igneous event intruding a future repository at Yucca Mountain, NV has previously been found to be above the regulatory threshold of 10^{-8}/year. Current regulations require that consequences for events with a probability greater than this be quantified. The potential consequences of such an igneous eruption on the proposed geologic repository at Yucca Mountain, and the extent of any radionuclide release resulting from such an event, has drawn considerable attention and analyses [3-4]. In general, these speculative consequence analyses have included multiple, compounded conservatisms that could result in relatively high predicted dose consequences.

To evaluate the amount of conservatism introduced by these assumptions, a series of best-estimate analyses regarding eruption conditions and magma interactions with engineered and natural barriers was conducted [1-2]. Building on these process-model analyses, a set of sensitivity analyses were made to explore the effect of alternative, more realistic assumptions on for predicted dose consequences. This approach is based on multiple lines of reasoning, rather than on unrealistic exclusion of relatively well-understood physical processes and barrier properties, and provides greater in-depth insight on the overall issue of risk from a possible igneous event at Yucca Mountain.

The overall goal is to place the igneous-event scenario, and specifically the extrusive (volcanic) release variant, into an appropriate perspective that more closely conforms to the regulatory philosophy established in EPA’s regulation 40 CFR 197 as a “reasonable expectation” of the events that might occur at the repository that the licensing of the repository will be based on a “reasonable expectation” of the events that might occur at the repository. To accomplish this, credible “reasonable” alternatives were presented [1], based on new analyses and additional literature information. In particular, specific consideration is made of on the contribution of the waste packages to limiting releases. By progressively calculating the relative contribution of multiple processes and barriers, it is
demonstrated that potential release from an extrusive igneous event are many orders of magnitude below regulatory limits for a repository at Yucca Mountain.

In a previous report [2], the various assumptions and sub-processes inherent in such a scenario for a repository at Yucca Mountain were examined. Specific steps were evaluated, including:

- probability of a future igneous event (basaltic dike) below Yucca Mountain,
- ascent and propagation of dike near the proposed site of the repository,
- magma-drift interaction,
- indirect magmatic effects on waste package containment (for intrusive-release pathway),
- magma-waste package interaction (for extrusive-release pathway),
- enhanced release and transport through the repository system (for intrusive-release pathway),
- dispersal of erupted contaminated magma (for extrusive-release pathway), and
- biosphere pathway analysis.

There is a long chain of events and processes that must occur for radiologically significant amounts of radionuclides to reach the compliance point from an igneous eruption at the Yucca Mountain site. As a result of the investigations and analyses [1], it has been concluded, based on multiple lines of evidence, that even in the unlikely event that an igneous event were to occur at Yucca Mountain, it is improbable that waste packages will be breached by the interactions of the magma during the active eruption period. Based on these findings for an igneous extrusive event within the Yucca Mountain repository footprint, the reasonable expectation consequence would be zero release of radioactive matter from the repository to the atmosphere during the extrusive event. It is further concluded that even in the extremely unlikely event that some radionuclides were released to the atmosphere as a result of such an event, only a negligibly minute fraction of the material could be transported to the regulatory compliance point. Even in this case, the conditional dose estimates would be many orders of magnitude below the regulatory limit.

These conclusions are based on multiple and parallel lines of evidence. The most important of these lines of evidence are as follows:

- Conditions at the drift level are less extreme than has been assumed previously by the NRC and DOE. This conclusion is based on examination of analogous volcanoes in the Yucca Mountain region. Prior analyses by DOE and NRC have included conditions from worldwide eruptions that were considered to be analogous to an eruption that might occur at Yucca Mountain. However, these conditions have not been observed at volcanic sites in the Yucca Mountain region despite intensive investigations. Limiting the potential behaviour of an assumed volcanic event at Yucca Mountain to that observed in the region significantly decreases the potential intensity of the eruption.
- Magma entering the drifts of the Yucca Mountain repository can be assumed to behave in a manner similar to its behaviour at the ground surface, which is less violent than has been assumed in the analyses of DOE and NRC to-date. Examination of lava flows in the Yucca Mountain region shows that the initial dike propagation will be characterized by effusive flow and lava fountains, progressing to a Strombolian stage.
- Conclusions drawn by Woods et al. [3] regarding potential drift-magma interactions and potential flow paths for magma within the repository have been discredited both by the Igneous Consequences Peer Review (ICPR, an independent review panel commissioned by DOE) Panel [4] and by EPRI contractors’ model results reported herein. Magma-drift interactions are far less severe than those postulated by Woods et al. [3]. The only credible ascent mechanism for the dike is continued propagation along the original line of ascent as opposed to the “dog-leg” scenario proposed by Woods et al. As the eruption progresses to a Strombolian stage, the only waste packages that could conceivably contribute radionuclides to an extrusive release would be those that were originally located directly in the magmatic vent as it arose. Furthermore, based on more recent and appropriate analytical methods [1, 2], the pressure wave may be established
within the repository as a result of the intersection of the magmatic dike with a drift is shown to be significantly less forceful than that postulated by Woods et al. [3].

- The waste package provides a very significant barrier to release of radionuclides during the scenario. This barrier has heretofore been neglected in prior analyses carried out by DOE and NRC. Consideration of credible failure mechanisms leads to reasonable expectation that no waste packages will fail during a postulated igneous eruption event at Yucca Mountain.
- Any magma entering the drifts and flowing away from the vent can be expected to cool and solidify in a manner that, in effect, will isolate the remaining waste packages from the magmatic dike that flows to the surface.

2. Conditional cases

Additional lines of evidence to support EPRI’s conclusion that the dose consequences due to an igneous eruption through the repository are negligible were explored using sensitivity analyses [1]. These sensitivity analyses explore the implications of a series of conservative assumptions used by DOE and NRC in their past analyses. It was shown that a series of compounding conservatisms used by DOE and NRC lead to overestimates of about nine orders of magnitude associated with any potential radionuclide dose that may be received by a member of the public at the regulatory point of interest. This result is extremely conservative. However, even with all of these conservatisms, the DOE analysis of the igneous extrusive scenario is still in compliance with the applicable regulations. Any potential changes to the analysis to introduce more reasonable behaviour into the DOE model would significantly decrease the calculated doses.

3. Summary

As a result of the investigations and analyses, it has been concluded, based on multiple lines of reasoning, that in the unlikely event that an igneous event were to occur at Yucca Mountain, it is probable that waste packages will not be breached by the actions of the magma during the active eruption period, with no expected release of radioactive material from the repository as a result of a volcanic event. Based on these findings, the expected consequence of an igneous extrusive event within the Yucca Mountain repository footprint would be zero releases of radioactive matter from the repository to the atmosphere during the extrusive event. It is further concluded that even in the extremely improbable event that some radionuclides were released to the atmosphere as a result of such an event, only a negligibly minute fraction of the material could be transported to the regulatory compliance point, and that dose impacts would be many orders of magnitude below the regulatory limit.

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Treatment of uncertainties over very long time periods in safety assessment

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Abstract. A recent court ruling in the United States has required the US Environmental Protection Agency (EPA) to modify its regulation for disposal of high-level waste and spent nuclear fuel at Yucca Mountain to take into account time periods greater than 10 000 years in the safety assessment. In this paper, the implications of this change to Yucca Mountain safety assessments are discussed, and issues associated with extending the time frame are presented. A stylized approach for the post-10 000 year period is recommended that would be protective of the public and implementable in the US legal and regulatory systems.

1. Background

In 2001, the U.S. Environmental Protection Agency issued a regulation for establishing the safety of the proposed repository for spent nuclear fuel and high-level waste at Yucca Mountain, Nevada, USA. In keeping with requirements established by the U.S. Congress, the regulation was based upon and consistent with the recommendations of a Committee of the National Academy of Sciences (NAS) \cite{1}. The resulting regulation was challenged legally by several organizations on thirteen different aspects of the regulation. In July 2004, the regulation was upheld in U.S. District Court on all aspects except for the time frame of the analysis. The Court ruled that the 10 000-year time cut-off for quantitative comparison of safety assessment results with the criteria was not consistent with the recommendations of NAS \cite{1}; quantitative calculations should be carried out to peak dose. The EPA is currently in the process of issuing a new regulation that would accommodate the Court’s ruling.

This paper examines technical elements of carrying out safety assessments into the distant future. Approaches are proposed to address uncertainties associated with the distant future that would be implementable in US legal and regulatory systems.

2. Stylized approaches.

Among the most important technical approaches in the literature for dealing with uncertainties at long times is the use of “stylized approaches.” This term is endorsed throughout the literature, both in the USA (40 CFR 197, 10 CFR 63) and internationally \cite{2, 3, 4, 5}. Surprisingly, however, there is not an accepted definition in the literature of the meaning of this term. Consequently, before addressing the use of stylized approaches, a working definition of stylized approaches is established and described here.

A stylized approach is defined as:

A set of assumptions established by policy that is used to limit the range of uncertainties considered in the safety assessment, while retaining the ability of the assessment to meaningfully evaluate the ability of the repository to protect public health and safety.

Some uncertainties, most prominently those associated with human actions, are accepted to be unquantifiable from a technical perspective. The stylized approach is used to establish bounds on the behaviour of people in the future. Stylized approaches are invoked to limit speculation on what might happen in the future, so that the safety assessment may focus on issues that are of greater importance in establishing reasonable assurance of regulatory compliance. So, for instance, EPA has established a
stylized approach in establishing the Reasonably Maximally Exposed Individual (RMEI) and the characteristics to be considered in RMEI behaviour.

Stylized approaches may also be invoked to limit the Features, Events, and Processes (FEPs) considered in the performance assessment. For instance, NAS [1] called for the use of a stylized scenario for inadvertent human intrusion that establishes the processes and exposure pathways to be considered in the scenario. Similarly, a number of European programs consider near-surface processes to be so uncertain beyond 10 000 years that they move away from dose or risk criteria to other indirect indicators of system safety [5, 6].

Stylized approaches are important in differentiating between scientific uncertainty, which may be large and growing at all times, and uncertainty in the regulatory decision, which must be manageable at all times. As discussed earlier in this chapter, these two concepts need to be kept clearly differentiated. The use of stylized approaches is an important tool in achieving balance between the two. Hence, in situations in which scientific uncertainty is large and growing, it may be appropriate to use stylized approaches to manage those uncertainties in order to focus on issues paramount to assurance of safety. It is important that application of a stylized approach should encompass reasonable bounds on system behaviour, so that the uncertainties are managed, not ignored.

3. Application to Yucca Mountain

A stylized approach appropriate for Yucca Mountain has several elements. First, regulatory and legal constraints require the use of dose or risk at all times in the assessment [1]. Second, a stylized approach to unpredictable future human behaviour [1] has already been adopted in the current regulation; this aspect of the regulation was not vacated by the Court, and remains in force. Third, the unpredictability of climate beyond about 10 000 years results in additional uncertainties in human behaviour; similarly, human behaviour may influence climate in an unpredictable way.

For the sake of internal consistency in the analysis, it is therefore recommended that a stylized approach to compliance model development be taken for time periods beyond 10 000 years. In particular, a fixed set of climate scenarios should be adopted. At most, only two scenarios are necessary to reasonably “bound” the effects of future climate: present-day “interglacial” and “glacial”. This is because the present-day climate results in the maximum human use of groundwater while the “glacial” climate would likely result in the maximum groundwater flow through the repository and to the compliance point. Given the surface water that would likely exist during a “glacial” climate in the Yucca Mountain vicinity, it is reasonable to assume that groundwater use would be dramatically lowered. Furthermore, given the large amount of dilution of any contaminants in the ground and surface water that would likely occur during the “glacial” climate, simply assuming the present-day “interglacial” climate exists for the entire duration of the compliance period would result in a reasonable upper bound on the effect of climate on peak dose risk.

The recommendation to establish the climate state in the compliance model to the present-day interglacial is based on the following considerations:

- recent evidence suggests that net infiltration over time periods spanning multiple climate states has been more constant than previously understood;
- biosphere analyses for the present-day interglacial climate are reasonably bounding due to the relatively high use of groundwater and higher atmospheric dust loadings;
- the goal of maintaining an internally consistent compliance assessment requires that future human behaviour be consistent with changes in the surface environment in different climate states. It would be impossible to avoid having to make largely arbitrary assumptions about such future human behaviour as it doesn’t exist in the Yucca Mountain region today; and
- the only climate state for which more detailed information is available upon which to develop and defend net infiltration and biosphere models is the present-day climate.
A number of potential options exist regarding how to implement regulations in the post-10 000-year period. Key considerations in promulgating a regulation involving compliance periods in excess of 10 000 years that:

- are consistent with the Court ruling;
- result in a “meaningful” standard that protects public health and safety in a constructive and equitable manner; and
- are “reasonable” and would be implementable in a regulatory environment.

Here, "implementable" means that during licensing the NRC would be able to determine that appropriate information and analyses were presented upon which NRC could make a compliance determination for such a long compliance period. A “meaningful” regulation should require that the conceptual models in the compliance assessment be internally consistent, in which all parts of the analysis are significant to real potential consequences that may arise from the repository, and which lend themselves to clear understanding of the implications of the analysis to the regulatory decision. This argues for practical, clear boundaries to be established in the regulation to avoid undue speculation about unusual conditions that are not central to the performance of the facility.

Features that should be addressed in the development of the new Yucca Mountain standard:

- The regulation needs to recognize that, as a result of the increase in uncertainty, particularly associated with climate, after 10 000 years, there is a need to stylize conditions for the period after 10 000 years. The current regulations stylize human behaviour, limiting the range of conditions that need to be incorporated into the analysis. Furthermore, international safety regulations also implicitly recognize and implement needed stylized safety standards for periods beyond 10 000 years because of climate change. In a new regulation with an extended regulatory time period, a comparable level of stylization is appropriate for establishing bounds on climate, and on FEPs that need to be considered in the analysis. These additional stylizations are necessary, since the uncertainties of climate and human behaviour in the far future introduce an arbitrariness to the analysis while not increasing public protection.

- Steady-state climate conditions should be applied for the duration of the analysis. This would avoid the conceptual and computational difficulties in evaluating system behaviour during transients between climate states – especially when uncertainties in the “details” of those future transitions are large. Simply assuming present-day interglacial climate continues until the time of peak dose would be no less arbitrary than alternative assumptions. This approach is preferred as it avoids the need to make largely arbitrary assumptions about the details of how a future climate state would be expressed (in terms of net infiltration, human behaviour, and other biosphere changes) at the Yucca Mountain site.

- The increases in uncertainty at long times as well as issues of balancing intergenerational and intragenerational equity argue for a relaxation of the current stringent 15 mrem/y dose constraint for the time period beyond 10 000 years. From a purely technical perspective, and in harmony with international recommendations on radiation protection, a dose constraint on the order of 100 mrem/y is considered to be protective of public health and safety in uncontrolled circumstances [7].

- FEPs to be retained in the analysis should be based on the current standard of $10^{-4}$ probability during the first $10^4$ years, with an allowance for omitting FEPs that are judged to have low consequence to the analysis. In essence, this approach supports the concept that the current FEP screening, based on the initial 10 000 year time period, is acceptable (and conservative) as the basis for the longer-term analysis. Alternative arguments are possible using the concept of Negligible Incremental Risk, which show that the $10^{-8}$ per year probability cut-off is extremely conservative compared to negligible dose and risk values. Given increased uncertainties in the long time periods, credible identification of FEPs with probabilities of occurrence as low as $10^{-8}$ per year will be difficult.
REFERENCES


Abstract. A safety assessment code, SAGE (Safety Assessment Groundwater Evaluation), has been developed to describe post-closure radionuclide releases and potential radiological doses for low- and intermediate-level radioactive waste disposal in an engineered vault facility. The code has been subjected to a variety of verification and validation tests. One of these tests is presented, which is compared against safety assessment results published from IAEA ISAM Vault Safety Case, to benchmark the reliability of system-level conceptual modelling of the code. SAGE compares well with existing analyses and is anticipated to give reliable results when applied for the safety assessment of a near surface disposal facility in Korea.

1. Introduction

Based on the steps in the IAEA methodology for the safety assessment of near-surface disposal facilities, SAGE (Safety Assessment Groundwater Evaluation) has been developed as an integrated system-level safety assessment tool. It has been developed to provide functionality in evaluation releases to groundwater as part of a total safety assessment for development and licensing of an engineered concrete vault disposal facility [1].

As a part of the IAEA co-ordinated research project on Improvement of Safety Assessment Methodologies for near surface disposal facilities (ISAM), Vault Safety Case (VSC) was developed to test and understand safety assessment approaches and to provide practical experience in their implementation [2]. In this paper, features and applications of the SAGE are presented to be compared against the existing safety assessment results of VSC and to benchmark the reliability of system-level conceptual modelling of the SAGE.

2. Development of a system-level safety assessment tool

The overall scheme of the code for safety assessment is shown in Fig. 1. The compartment approach is implemented in the code for near field, far field, and biosphere modelling. SAGE is organized to produce an overall system-level model for a low- and intermediate-level waste (LILW) disposal facility. SAGE has a modular structure, and has the capability to treat input parameters either deterministically or probabilistically. The code has been written to easily interface with more detailed codes for specific parts of the safety assessment. In this way, the code’s capabilities can be significantly expanded as needed. The graphical user interface (GUI) is linked to a database for managing site-specific data. This function is implemented in an input database/quality assurance module for safety assessment, named QUARK (Quality Assurance and database for Radioactive waste management in Korea), which interacts with SAGE [3]. The functionality of SAGE has been tested methodically against a number of requirements established for its use. In undertaking these tests, essentially all of the technical capabilities of SAGE have been benchmarked against other analyses [4]. These benchmark tests comprise four types, i.e., near-field model tests, far-field model tests, biosphere models tests, and system-level tests. These tests provide a high degree of confidence that the code has been implemented correctly, and that the results generated using the code have a high degree of reliability.
3. IAEA-ISAM vault safety case

To benchmark SAGE against an existing system-level analysis, the ISAM VSC, the radiological impacts of radionuclides on a critical human group have been evaluated for a hypothetical near-surface radioactive waste disposal facility in Vaalputs, South Africa. The analysis deals with a groundwater release scenario, which corresponds to the use of contaminated water in the biosphere, after migration of the radionuclides through the geosphere. The water and contaminant transport is associated with failure of the cover, which exposes waste to the infiltrating water. Due to the water flow through the engineered structure of vault disposal system, the contaminated water flows downward from the waste matrix and the rest of the vault to the far field (the unsaturated layer and aquifer). The aquifer is assumed as the only source of biosphere contamination. The parameters for the modelling of source-term, geosphere and biosphere are mainly obtained from the site-specific data. Table I lists the radionuclide inventory at facility closure and decay chains considered in this safety assessment. All of the parameters necessary for the comparison are given by IAEA [2].

It was assumed that the near-field barriers degrade with time. The closure cover is assumed to be maintained during the 100-year active institutional control period, but then starts to degrade so that it allows 50 % of the total precipitation to pass from 100 to 500 years, and it no longer limits the rate of water infiltration after 500 years. The near-field barrier is also assumed to be degraded chemically after 500 years. In order to consider changing the distribution coefficient Kd's in the vault as a function of time, SAGE model assumes that Kd for each element and in each material in the near field is a piecewise step function in time.

For flow in the unsaturated zone (UZ), it was assumed that there are a series of fractures below the disposal facility and that an equivalent porous medium approximation can be made to represent flow and transport in the zone. In the saturated zone (SZ), it is assumed that the water that has percolated down from the UZ is intercepted by a series of fractures. It was assumed that the series of fractures in the SZ that intercept the percolating water from the disposal facility can be represented by a single streamtube.
Table I: Radionuclide inventory at facility closure and decay chains

<table>
<thead>
<tr>
<th>Nuclides</th>
<th>Inventory (Bq)</th>
<th>Decay Chain</th>
</tr>
</thead>
<tbody>
<tr>
<td>H-3</td>
<td>1E+15</td>
<td></td>
</tr>
<tr>
<td>C-14</td>
<td>1E+13</td>
<td></td>
</tr>
<tr>
<td>Ni-63</td>
<td>1E+10</td>
<td></td>
</tr>
<tr>
<td>Sr-90</td>
<td>1E+15</td>
<td></td>
</tr>
<tr>
<td>Te-99</td>
<td>1E+14</td>
<td></td>
</tr>
<tr>
<td>I-129</td>
<td>3E+10</td>
<td></td>
</tr>
<tr>
<td>Cs-137</td>
<td>6E+09</td>
<td></td>
</tr>
<tr>
<td>U-234</td>
<td>8E+15</td>
<td>Th-230→Ra-266→Pb-210</td>
</tr>
<tr>
<td>U-238</td>
<td>5E+10</td>
<td>U-234→Th-230→Ra-226→Pb-210→Po-210</td>
</tr>
<tr>
<td>Pu-238</td>
<td>2E+10</td>
<td>U-234→Th-230→Ra-226→Pb-210→Po-210</td>
</tr>
<tr>
<td>Pu-239</td>
<td>3E+10</td>
<td>U-235→Pa-231→Ac-227</td>
</tr>
<tr>
<td>Am-241</td>
<td>2E+10</td>
<td>Np-237→Pa-233→U-233→Th-229</td>
</tr>
<tr>
<td>TOTAL</td>
<td>1E+16</td>
<td></td>
</tr>
</tbody>
</table>

4. Results from system-level test: ISAM vault safety case

The results from the SAGE calculation have been compared with those from other computer codes for the same facility and scenario. Despite small differences in the conceptual models underlying the three analyses of the problem, good agreement is achieved in comparing peak releases into the aquifer. In particular, it was observed that minor changes in flow rate through the disposal facility produced significant changes in the results. A comparison of prior analyses in the literature with SAGE results is shown in Table I for releases into the aquifer, and in Table II for well concentrations. Particularly good agreement is found between the results of Little [5] and those of SAGE. This agreement is the result of rather close conceptual and mathematical models used in the two analyses. Key differences in the mathematical models would come from the assumptions on changes in the infiltration of precipitation into the repository and on changes in the physical status of the concrete in the repository from non-degraded to fully-degraded. Little used linear relationships for infiltration and concrete failure modelling. Nevertheless, it was observed that the results were broadly the same because of the relatively rapid assumed transit time of water along the fracture and the low distribution coefficients for the radionuclides in the geosphere. By contrast, Kim [5] used slightly different assumptions, consistent with the use of his model.

Table II: Comparison of peak flux rates to the aquifer from published values and SAGE

<table>
<thead>
<tr>
<th>Nuclides</th>
<th>Kim (KHNP-NETEC)</th>
<th>Little (Quintessa)</th>
<th>SAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Time (yr)</td>
<td>Peak Flux (Bq/yr)</td>
<td>Time (yr)</td>
</tr>
<tr>
<td>C-14</td>
<td>31000</td>
<td>3.3E+5</td>
<td>30000</td>
</tr>
<tr>
<td>Te-99</td>
<td>2500</td>
<td>1.8E+7</td>
<td>2500</td>
</tr>
<tr>
<td>I-129</td>
<td>6900</td>
<td>1.3E+6</td>
<td>8000</td>
</tr>
<tr>
<td>U-234</td>
<td>65000</td>
<td>5.2E+5</td>
<td>40000</td>
</tr>
<tr>
<td>U-238</td>
<td>65000</td>
<td>5.2E+5</td>
<td>50000</td>
</tr>
</tbody>
</table>
Table III: Comparison of peak concentrations in the well from published values and SAGE.

<table>
<thead>
<tr>
<th>Nuclides</th>
<th>Kim (KHNP-NETEC)</th>
<th>Little (Quintessa)</th>
<th>SAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Time (yr)</td>
<td>Peak Concentration (Bq m(^{-3}))</td>
<td>Time (yr)</td>
</tr>
<tr>
<td>C-14</td>
<td>31000</td>
<td>6.7E+1</td>
<td>30000</td>
</tr>
<tr>
<td>Tc-99</td>
<td>2500</td>
<td>3.7E+3</td>
<td>2500</td>
</tr>
<tr>
<td>I-129</td>
<td>7000</td>
<td>2.7E+2</td>
<td>8000</td>
</tr>
<tr>
<td>U-234</td>
<td>73000</td>
<td>1.3E+2</td>
<td>50000</td>
</tr>
<tr>
<td>U-238</td>
<td>59000</td>
<td>1.1E+2</td>
<td>50000</td>
</tr>
<tr>
<td>Ra-226</td>
<td>120000</td>
<td>2.8E+0</td>
<td>100000</td>
</tr>
<tr>
<td>Pb-210</td>
<td>120000</td>
<td>9.3E+0</td>
<td>100000</td>
</tr>
<tr>
<td>Po-210</td>
<td>120000</td>
<td>4.0E+0</td>
<td>100000</td>
</tr>
</tbody>
</table>

5. Conclusions

This paper described a new computer code, SAGE, developed as a tool for use in safety assessments of the Korean concept for LILW disposal. SAGE has been developed to provide functionality in evaluation releases to groundwater as part of a total safety assessment for development and licensing of an engineered concrete vault disposal facility. A discussion of the verification of SAGE compared to a system-level safety assessment analysis for a groundwater release scenario in IAEA VSC has been provided in this paper. SAGE results have been compared against those from the other computer codes to ensure that the implementation is correct. The functionality of SAGE has also been tested. SAGE compares well with existing codes for conducting LILW safety assessments, and is anticipated to give reliable results when applied for the assessment of near-surface disposal facilities in Korea.

Acknowledgement. This work has been carried out as a part of the Nuclear R&D programme funded by the Ministry of Science and Technology in Korea.

REFERENCES


Abstract. In Malaysia, radioactive materials are mostly used in the industrial, medical, research and educational sectors which will inevitably generate radioactive wastes. As a major tin mining nation Malaysia has been generating radioactive wastes that are now known as Technologically Enhanced Radioactive Materials (TENORM). In this paper, the role and responsibilities of the Atomic Energy Licensing Board (AELB) is discussed with attention given to current practice and regulatory infrastructure aspects on radioactive waste management in Malaysia.

1. Introduction

Atomic energy has been used in Malaysia since 1897 when a hospital in Taiping, Perak used X-ray machine for medical purposes. The atomic energy legislation concerning radioactive materials was issued in 1968 and known as Radioactive Substances Act 1968. Due to rapid development of atomic energy activities in Malaysia which requires more effective control, inspection and enforcement, the Atomic Energy Licensing Bill was drafted. This Bill was passed by Parliament in April 1984 as the Atomic Energy Licensing Act (Act 304). Atomic Energy Licensing Board (AELB) was established on 1st February 1985 after Act 304 was gazetted on 24th June 1984. AELB’s main objective is to regulate and control all activities concerning atomic energy and to protect radiation workers, members of the public and the environment from radiation hazards. As stated in Act 304, AELB merely controls the use of atomic energy in industries except medical industries. The regulatory body responsible for supervising and controlling the use of atomic energy in medical industries is the Ministry of Health.

Presently, there are three main sources of radioactive wastes in Malaysia which are:

(a) waste generated from industrial, medical, research and education sectors
(b) TENORM waste resulting from mining, oil and gas industries
(c) spent fuel from research reactor

AELB is responsible on establishing control over radioactive wastes generated by users in industrial, research and education sectors. This paper presents some consideration on national legislative and regulatory system covering the aspects of radioactive waste management.

2. Review of the legislative and regulatory framework

Under the present legislation, management or disposal of radioactive wastes is specifically dealt with under Sections 26 – 31 of Act 304. These sections empower AELB to ensure licensee to obtain appropriate license for activities (accumulate, transport, dispose) concerning radioactive waste and to take actions on activities which are deemed unsafe. However, these provisions do not provide details on measures that should be taken by licensees. Act 304 and its relevant regulations [1] provide requirements on annual dose limits that need to be adhered to by licensees or users in any activities involving radioactive materials but do not elaborate on specific requirements needed to ensure safe management of radioactive waste. Furthermore, Act 304 and its relevant regulations do not restrict any individual or organization to apply for a license to manage radioactive waste.
The current legislation only covers activities related to management of radioactive waste in a general manner and there is no specific provision on the disposal of radioactive waste. As such, AELB has adopted an interim policy where radioactive waste shall be managed, but not disposed of into the environment:

(a) kept in storage by the licensee or user
(b) returned to supplier e.g. disused sealed sources
(c) sent to National Radioactive Waste Management Centre, Malaysian Institute for Nuclear Technology Research (MINT).

3. Establishing a national policy

Presently, there is no policy established for the safe management of radioactive waste. In 1997, special committee has been set up with the task of formulating a national policy that deals with the safe management of radioactive waste in Malaysia. This committee consists of various groups from operating organizations, non-governmental organizations and those who may have interest in the issue.

A draft policy has been submitted to the Ministry of Science, Technology and Environment for comments and approval in early 1999. The draft policy basically adopts the IAEA’s nine Principles of Radioactive Waste Management [2] and contains strategies and action plans to be taken when implementing the policy. A more holistic approach towards the management of waste is being studied and this will determine whether the draft policy on radioactive waste will be part and parcel of this.

4. Current practice

Since there is no specific regulation on disposal and management of radioactive wastes, AELB enforces its interim policy through licensing procedures and conditions to its licensees or users.

The issue concerning a closed mineral processing plant has necessitated the development of a new regulation that deals with the safe management of radioactive waste. The draft regulation has been submitted to the Ministry of Science, Technology and Innovation for review and approval.

5. Conclusion

The policy for safe management of radioactive waste has yet to be established. AELB is in the process of developing a new regulation regarding safe management of radioactive waste while adopting an interim policy in dealing with issues concerning radioactive waste at the present time.

REFERENCES


Romanian national repository for low and intermediate level radioactive waste, Baita, Bihor county (DNDR): Present status and further developments

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Abstract. The proper application of the nuclear techniques and technologies in Romania started in 1957, once with the commissioning of the Research Reactor VVR-S from IFIN-HH-Magurele. During the last 40 years, thousands of nuclear application units appeared with extremely diverse profiles (research, biology, medicine, education, agriculture, transport, all types of industry) which used radioactive sources in their activity and produced nuclear wastes. The Radioactive Waste Treatment Plant (STDR) from IFIN-HH was committed in collaboration with companies from the United Kingdom and became operational in 1975, being the only authorized and specialized institution for the management of the non-fuel cycle radioactive wastes from all over Romania. In 1985 the National Repository for Low and Intermediate Radioactive Waste (DNDR) – Baita, Bihor county, was built and put in operation. It is sited in the Apuseni mountains, in an old exhausted uranium mine. The facilities were designed and built according to the philosophy of the 1960s and 1970s, common to ex-socialist countries, except for some West-European imports. The paper describes the current operational experience at the National Repository for Low and Intermediate Level Waste – Baita, the facility, further developments by way of revamping of existing processes equipment and systems in order to enhance the safety.

1. Introduction

The site selection was based on preliminary studies concerning the geology, hydro-geology, seismic, meteorological and radioactivity of the area, and also on mining technical studies.

The site is situated in the Northwest Carpathian Mountains, about 8 km from the nearest community, the Baita village and about 10 km from the small town of Nucet.

The Avram Iancu uranium deposit (which include galleries 50 and 53) was found by Russians in 1952 using gamma prospecting, being considered one of the most important uranium deposits from Romania. In time, the activity was diminished and now the exploitation is almost shut down. The repository area is totally exhausted with no economic potential and is planned to start a re-ecologisation and restoration project in the whole area. The repository was originally an exploration drift with several galleries at an open pit uranium mine, which was exhausted in the early 80's. Mining was carried out by blasting. This operation created many fractures in the rock, which now might constitute preferential pathways for water to penetrate into the repository. The repository consists of chambers excavated transversally to the central gallery.

Waste packages (mostly 200 litre-drums) are brought to the disposal facility by a 16-ton truck. Then, using forklifts, drums are stacked inside the disposal chambers. Stacking is performed in alternate rows of 6 and 5 drums up to the top of the chamber. There are usually 25-30 drums per stack.

Since 1995, over each row, bentonite powder is poured on the drums. This is to absorb possible penetrating water and to fix caesium. In order to prevent bentonite to flow outside the drums, a wood panel is erected progressively with the stacking. Now three chambers are full.

Closure of the chambers is performed by bricks. There are in total more than 6700 drums already placed in disposal chambers. This corresponds to approximately one third of the disposal capacity of the existing facility. Additional galleries could be dug in the future extending thereby the disposal capacity up to 200 000 drums.
One of the main reasons for selecting the Baita Bihor site is related to the unsaturated (with water) character of the surrounding rocks. However, due to rainfall, some water sometimes flows from the walls of the repository notably in the access gallery.

In principle, the water that could penetrate into the disposal chambers could be collected in a drainage tube covered with rugged concrete plates. At the exit of the facility there is a tank collecting all water. Measurement of the activity is carried out once every six months. So far no significant activity was found in the drainage system.

2. Licensing of disposal operations at Baita Bihor

According to the law on the safe deployment of nuclear activities, the National Commission for Nuclear Activities Control (CNCAN) is the regulatory and licensing body with responsibility for nuclear safety, radiation protection, decommissioning and radioactive waste management. In this context, the Baita Bihor repository for institutional radioactive waste is regulated by CNCAN in conjunction with the Bihor county authorities and the Municipality of the local town Nucet. The licence is actually based on older authorisations, which were quite restrictive in terms of specific activity limits for waste packages as well as for the total inventory of radionuclides that can be accepted by the disposal facility.

It is worth noting that these limits were based on very rough safety assessment. Therefore in the late nineties, CNCAN was interested in performing a safety case for the site that could at least justify the limits defined earlier and if possible make them more flexible with respect to a number of key-radionuclides.

This action led to the implementation of a PHARE funded study entitled "Preparatory measures for the long-term safety assessment of the low-level radioactive waste repository Baita Bihor" from September 2000 to September 2001. However this project only enabled the CNCAN and IFIN-HH experts to familiarise themselves with the methodology for safety assessment. No dose calculation was performed during the period covered by the contract. However, during the first half of 2002, IFIN-HH experts succeeded to preliminarily quantify a first set of doses for the critical groups of the population. Based on the results of that study, CNCAN has just granted a new licence to IFIN-HH for the operation of the Baita Bihor repository up to 2009. The new licence enables the disposal of waste packages with higher specific activities for Tc-99, I-129, Cl-36, Co-60 and Cs-137. Meantime, IFIN-HH will have to prepare a Preliminary Safety Assessment Report (PSAR) within the framework of a Phare project, under development now, having the completion deadline in October 2006.

3. Operation of DNDR-Baita

Disposal and ventilation galleries are dug up in flisoid formation which has maximum thickness in this zone. The main types of rock met in galleries are characterized by low porosity and humidity, which practically means that they are dry and compact. From the mechanical characteristics point of view, they have big resistance at compression (f=12.5). Thus, their natural stability suggests that for almost 25 years from excavation, with small exceptions, the galleries maintain the initial profiles. The main water leakage results from precipitation. This fact as well as the situation of the geologic basement eliminates the existence of a hydrostatic level. The absence of permanent water sources, assure a low inundation level, practically null, taking also in consideration that the above rock package has 160-180 m thickness. The hydro-geologic prospective performed underground as well as the chemical analyze of the water conducted to the result that groundwater does not origin from underground springs. Hence the rainfall and melted snows infiltration water (through fractures) are the single sources of groundwater.

The planning was carried out taking in consideration the total length of the galleries and the annually deposited drums, obtaining an optimum profile of 10,5 m. Locally the walls were plastered with Portland cement selected concerning its small alkalinity and aggressiveness to water. The same type of cement is now used in the radioactive waste conditioning process.

In the technological disposal process bentonite, wood and cement brick is used. Bentonite is used as backfilling material and engineered barrier, taking into consideration its very good plasticity and
absorption capacity, which diminishes the radionuclide migration possibility from deposited drums. Between the drums ranges, wood shuttering is placed. When a gallery is filled up, it is tightened with cement bricks. These materials are placed near the working area, inside the gallery.

The National Institute for Physics and Nuclear Engineering “Horia Hulubei”, pursues in a dynamic way, by radioactivity measurements and potentially radioactive elements migration, the efficacy of the applied methods regarding to assure the nuclear security of the population and the environment.

4. Up-grading of DNDR

It must be underlined that operation of the Baita Bihor repository started despite the fact that it was not fully equipped. Therefore, after nearly 20 years of operation, its rather weak equipments are largely degraded and upgrading has become more and more urgent, especially in the context of the next dismantling operations of the VVR-S research reactor at Magurele that will generate rather high volumes of radioactive waste to be disposed of.

The replacement of process equipment and control instrumentation is imposed by factors such as:
1. damage during operation;
2. up-dating of operational conditions;
3. meeting modern standards.

Shortly, modernization of DNDR is foreseen in order to improve the safety of disposal operations of institutional radioactive waste at Baita-Bihor:
- construction of a one way asphalt road with intersection points and drainage system;
- at present, the access road creates many problems concerning the normal operation of the repository. Because of the meteorological conditions (large quantities of precipitation which produce deep erosion processes) the road cannot be used from November until April, and in every spring must be levelled and filled up with rocks;
• a technologic building sited on the entrance platform which will include rooms for ventilation and electric equipment, laboratories, a hall for unload the drums and a space for storage until their transfer in gallery, offices, rest room, others;

• replacement of the existing electric and ventilation systems; From 1985 are used the same electric and ventilation equipment, without any capital overhauling. Because of the high humidity they are damaged and require frequent interventions for local remedies;

• replacement of the drainage system; Generally, the drainage system will be based on the same philosophy being foresight to rebuild the water collection channels and the flagstones;

• waterproofing of the disposal and transport galleries; Because in the area were done shootings for uranium ore exploitation, in time appear cracks in the disposal and transport gallery;

• realization of a modern physical protection and communication system; The system will assure the physical protection during the whole disposal technologic process at DNDR;

• improvement of the radiation protection system, developed in accordance with the optimization and individual dose limitation principles.

5. Conclusions

Over the last two years, several systems were significantly improved: the electric and ventilation systems, the transport and manipulation system, the communication system, the physical protection system and the radiation protection system. By the end of 2005, a modern radiation protection and monitoring system and a suitable transport system (according to the road conditions) will be available. As of January 2004, a very important project for the Baita repository was undertaken: preparation of the Preliminary Safety Assessment Report, which is assumed to be completed in October 2006.
Self disposal of uranium sludge waste by reuse

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Republic of Korea

Abstract Uranium-containing wastes have been treated by electrosorption on a carbon electrode composed of activated carbon fibers (ACFs) in a continuous flow-through cell. Effective uranium (VI) removal is accomplished when a negative potential in the range of -0.3 to -0.9V (vs. Ag/AgCl) is applied to the carbon electrode. For a feed concentration of 100-350mg/l, the concentration of U(VI) in the cell effluent is reduced to less than 1mg/l. Electrosorption capacity over 552 mg uranium/g ACF is reached. The self disposal safety of the resulted liquid waste containing chemical salts and small amount of uranium was evaluated. Scenarios on exposure pathway by self-disposal of uranium-containing sludge were developed. Also, mathematical models were developed along each exposure pathway. Results of assessment of a reuse of uranium-containing sludge as a liquid fertilizer were as follows. Radiation effective dose of leafy vegetables showed the maximum value and their concentration limit on the waste liquid of uranium-containing sludge was 17.22 ppm. Radiation effective dose of grain showed the minimum value and its concentration limit on the waste liquid was 42.79 ppm.

1. Introduction

1.1. Sludge treatment

A variety of physical, chemical and biological methods have been proposed for the removal and recovery of uranium (VI) from contaminated water and waste streams. But, these techniques such as precipitation, coagulation, ion exchange and adsorption on activated alumina have been restricted in wide application due to their limited capacity, especially when the concentration of U(VI) in the waste water is relatively high [1]. Therefore, the research and development of the electrochemical method for the removal of high concentration uranium has been extensively investigated. Electrodeposition of metals on a variety of carbon materials such as glassy carbon, carbon foams was very effective [2]. For some metals like uranium, lead and strontium having a high reduction potential, electrodeposition is not a practical method. An alternative to electrodeposition is electrosorption, that is, adsorption of the metal cations onto a negatively charged carbon surface. The technique of electrosorption, which may use the electrical potential as the 3rd driving force to the traditional adsorption and ion exchange mechanism, has reversible characteristics of purifying waste solution by adsorption and concentrating contaminants by desorption. Carbon materials satisfy the basic requirements for an efficient electrode material, and have good radiation and chemical-stability. Especially activated carbon fiber (ACF), which can be easily made into a variety of types (textures or sheet), has a high specific surface area and electrical conductivity.

In this study, we conducted the experiments on a selective adsorption of uranium (VI) from high concentration chemical salt containing U(VI) to investigate the application feasibility of the electrosorption technique using ACF as a good conductive electrosorption adsorbent.

1.2. Self disposal safety evaluation

Contents of the waste liquid of uranium-containing sludge treated by electrosorption consist of salt such as NaNO₃, NH₄NO₃ and Ca(NO₃)₂ which is used as a nitrogen fertilizer or a material to protect soil acidity. The concentration of uranium in the waste liquid is reduced to less than 1ppm by electrosorption treatment. Accordingly, the waste liquid don’t contain radioactive hazard by uranium and can be reused as a nitrogen fertilizer through a simple chemical treatment. Documents on scenarios and concentration limit to meet the clearance level were investigated to perform the safety assessment for self-disposal of uranium sludge wastes. IAEA suggested 10μSv/year (=1mrem/year) to
individual as effective dose for deregulation [3,4]. Scenarios and concentration limit to meet the clearance level are established by safety assessment.

2. Experimental and self-disposal safety evaluation

2.1. Uranium sludge treatment

Flow-through adsorption experiments were carried out using a three-electrode electrochemical cell where electric current flows parallel to the solution flow. Working electrode (ACF) was placed on a platinum mesh (current collector). The counter electrode was platinum wire and the Ag/AgCl electrode was used as the reference electrode. All the potentials reported in this paper are relative to this reference electrode. The electro-chemical cell was connected with a potentiostat (EG&G Model 273). The fixed flow rates through the cell were controlled by a peristaltic pump. Various type of salts such as NaCl, NaNO₃, NH₄NO₃ and Ca(NO₃)₂ were tested as the supporting electrolyte.

![Diagram of uranium in liquid fertilizer](image)

**FIG. 1:** Exposure pathway on re-using uranium-containing sludge as a liquid fertilizer

2.2. Self disposal safety evaluation

When the waste liquid of uranium-containing sludge was sprinkled in farmland as a fertilizer, it was absorbed into crops directly or root of crops through soil like Fig. 1. Human being receives internal exposure by the ingestion of crops, inhalation, and external exposure. Scenarios on exposure pathway by self-disposal of uranium-containing sludge were developed. Mathematic models were developed.
along each exposure pathway. For assessment of a reuse scenario as a liquid fertilizer, each effective
dose by respiration, ingestion, and exterior exposure within 1 ha of landfill sprinkled with the waste
liquid of 1 ppm concentration was calculated by the developed models. Exposure by ingestion was
classified into leafy vegetables, root vegetables, fruits, and grains and effective dose of each exposure
was calculated with models.

3. Results and discussion

3.1. Sludge treatment

Electrosorption of U(VI) onto the ACF felt was carried out at various potentials. Effective U(VI)
removal is accomplished at all negative applying potentials, as shown in Fig. 2. When applying
potential ranges from –0.5 to –0.9V (vs. Ag/AgCl), up to 99.8% of U(VI) is removed from 350mg/L
feed solution by electrosorption and maintained throughout the test. However, in case of OCP (open-
circuit potential) the effluent concentration of U(VI) increased within 3h and finally reached the level
of the feed, indicating saturation of sorption capacity by the ACF. From these results, we confirmed
that the external negative potential exerted on the ACF electrode has great impact on the adsorption
capacity of the ACF. In a long-term test conducted with 350mg/L U(VI) feed at –0.9V, the effluent
concentration is maintained at less than 1mg/L, corresponding to a specific sorption rate of
1.68mg/(g • min). The cumulative uranium sorption within 16 h is about 552mg uranium/gACF. The
adsorbed uranium could be desorbed up to 92% for 10 h by passing a 1.0M NaCl solution through the
cell and applying a potential of +1.2V (Fig. 3).

![FIG. 2: Electrosorption of U(VI) with variation of potentials.](image)

![FIG. 3: Desorption of U(VI) at +1.2V](image)

3.2. Self-disposal safety evaluation

The result of the assessment of a reuse of uranium-containing sludge as a liquid fertilizer was shown in
Table 1. Radiation effective dose of leafy vegetables showed the maximum value and their
concentration limit on the waste liquid of uranium-containing sludge was 17.22 ppm. The radiation
effective dose of grain showed the minimum value and its concentration limit on the waste liquid was
42.79 ppm.
Table 1 Assessment result on fertilizer re-use scenario of uranium sludge wastes

<table>
<thead>
<tr>
<th>Crop Type</th>
<th>effective dose (applied 1ppm)</th>
<th>concentration limit (to meet 10 µSv/y)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Leafy vegetables</td>
<td>5.81×10⁻² mrem/yr</td>
<td>17.22 ppm</td>
</tr>
<tr>
<td>Root vegetables</td>
<td>4.44×10⁻² mrem/yr</td>
<td>22.52 ppm</td>
</tr>
<tr>
<td>Fruits</td>
<td>2.18×10⁻² mrem/yr</td>
<td>45.86 ppm</td>
</tr>
<tr>
<td>Grain</td>
<td>2.34×10⁻² mrem/yr</td>
<td>42.79 ppm</td>
</tr>
</tbody>
</table>

4. Conclusions

4.1. Uranium sludge treatment

It was found that Uranium in liquid Uranium waste containing hundreds ppm of U and thousands ppm of the salts such as ammonium nitrate and sodium carbonates can be effectively removed by electrosorption using pitch based activated carbon fibers to the level of less than 1mg/l which is enough to free release to the environments. Our study showed that the electrosorption process is a promising uranium decontamination technique by selective removing uranium in the lagoon sludge waste in an uranium conversion plant.

4.2. Self disposal safety evaluation

Result of assessment of a reuse of uranium-containing sludge as a liquid fertilizer was shown that radiation effective dose of leafy vegetables showed the maximum value and their concentration limit on the waste liquid of uranium-containing sludge was 17.22 ppm. While, Radiation effective dose of grain showed the minimum value and its concentration limit on the waste liquid was 42.79 ppm. Therefore uranium-containing sludge can be reused as a liquid fertilizer after concentration reduction of uranium in sludge by electrosorption.

REFERENCES

Suitability of incineration of NORM contaminated oil sludge wastes; Radiological impact assessment at a waste incineration and disposal site in Malaysia.

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\textsuperscript{b}School of Chemistry and Food Science, Faculty of Science and Technology, 
\textsuperscript{c}Safety Office, Universiti Kebangsaan Malaysia, Selangor, Malaysia

Abstract. The reprocessing of crude oil in Malaysia has resulted in large accumulation of oil sludge wastes contaminated with naturally occurring material (NORM) with various concentration of $^{238}\text{U}$, $^{232}\text{Th}$, $^{226}\text{Ra}$ and $^{40}\text{K}$. One of the common methods of disposal is through the storage of the oil sludge. This study was conducted to see whether incineration could be employed in the disposal of NORM contaminated oil sludge wastes and the exposure level of workers handling the oil sludge wastes. The specific activity of NORM was determined using HPGe-Gamma spectrometry system. Based on the available data, the projected radioactivity concentration and its radiological impact in the area involved were calculated using the RESRAD code. Results of in-situ radiation dose measurements at the storage area and on the oil sludge at a Crude Oil Terminal and at a Waste Management Centre in Malaysia were found to be between 0.10 – 0.20 $\mu$Sv/hr and 0.30 $\mu$Sv/hr respectively. Through the process of incineration at a temperature of about 1150 $^\circ$C, ash and slag generated were found to give a radiation dose measurements of between 0.20 – 0.35 $\mu$Sv/hr. Area and personnel radiation monitoring have given a readings of between 52.22 – 142.08 $\mu$Sv at various storage locations, while the readings for personnel monitoring were between 9.31 – 14.76 $\mu$Sv. The result shows that the specific radioactivity of $^{238}\text{U}$, $^{232}\text{Th}$, $^{226}\text{Ra}$ and $^{40}\text{K}$ after incineration at a temperature of 1150 $^\circ$C has shown no significant risk of external exposure to both personnel and environment. The RESRAD calculation shows that the projected maximum dose exposure to workers from $^{238}\text{U}$, $^{232}\text{Th}$, $^{238}\text{Th}$, $^{40}\text{K}$ and $^{226}\text{Ra}$ radionuclides at the disposal site was estimated to be about 2.055 x 10\textsuperscript{-2} mSv/yr for the first year of deposit of incinerated oil sludge waste, with a decrease to 1.328 x 10\textsuperscript{-2} mSv/yr for the 100 year period. The study has also shown that the potential risk to cancer at the site for the first year of about 4.232 x 10\textsuperscript{-5} decreasing to 3.715 x 10\textsuperscript{-5} for a period of 30 years. This suggested a suitability of using incineration method in disposing NORM contaminated oil sludge wastes.

1. Introduction

Naturally occurring radioactive materials (NORM) has been found to be present in the oil production and processing facilities since the early nineteen thirties. $^{238}\text{U}$ and $^{232}\text{Th}$, as well as $^{40}\text{K}$ present in the earth’s crust have been found to concentrate in ordinary scale and oil sludge deposits. Previous studies conducted by Heaton et al 1995 [1] and recently by Shawky 2001 [2] have shown an enhancement in activity levels of naturally occurring radioactive materials in the oil processing industry. A study conducted in the US have found that the averaged radionuclide concentration in oil sludge were between 2.1 kBq/kg for $^{226}\text{Ra}$ and 0.7 kBq/kg for $^{228}\text{Ra}$ [3].

One possible method of waste disposal suggested is by incineration [4]. In order to establish the effect of incineration, radiation dose measurements were performed during the whole process of transportation, incineration and storage of oil sludge and incinerated waste.

The incinerator consist of a rotary kiln, which was designed to process solid waste, high heating value liquid waste and aqueous waste. The normal operating temperature was about 1150$^\circ$C. Since very few studies have been conducted on incineration of norm-contaminated oil sludge, this study was done to look into the level of radiation dose during transportation, incineration and storage to area as well as to the personnel. The data gathered could be useful in formulating regulation in the disposal of norm-contaminated oil sludge [5].
In projecting the possible maximum dose expected in the process of disposing incinerated oil sludge waste, a radiological impact assessment was conducted using the established RESRAD (RESidue RADioactive) code developed at Argonne National Laboratory (ANL) in the United States.

2. Methodology

In situ radiation measurements were conducted at a Crude Oil Terminal in the East Coast and at a Waste Management Centre in Malaysia using portable survey meters. The area radiation monitoring at WMC was recorded using TLD 4500 Harshaw-Bicron with TLD100H as chipstrates.

The RESRAD calculation was done for the Incineration Centre and Disposal site. Base on the target i.e. the workers at the waste incineration and disposal centre working 6 days a week with an average 5 hrs outside. The centre is about 2 km away from housing areas. Four exposure pathways, inhalation, ingestion, external exposure were identified and used. The contaminated zone was an area of about 225 m² at a depth of 2.5 m, with a wind speed of 2.8 ms⁻¹ and an irrigation rate of 0.2 myr⁻¹. Data used were from the studies conducted while other data were from estimates and from Yu et al 1993 [6].

3. Results and discussion

From the study conducted, radiation dose at the Crude Oil Terminal was found to be between 0.10 – 0.20 µSv/hr while at the Waste Management Centre was about 0.30 µSv/hr. The equivalent doses for various locations in the complex recorded, were between 52.22 – 142.08 µSv/month. Similarly readings for personnel radiation dose were between 9.31 – 14.38 µSv. Also from the study conducted the radiation dose has slightly increases in the samples collected after incineration in comparison to the samples before incineration. Ash and slag recovered from the incinerator recorded readings of between 0.25 – 0.35 µSv/hr.

Results given in Table I, have shown that the specific activity for ²³⁸U increases from 25.3 Bq/kg to 97.3 Bq/kg. Estimates have shown that total activity of ²³⁸U decreases by 4.4 % compared to 13.9 % for ²³²Th after incineration. A much higher loss in ²²⁶Ra by about 34.3 % is because of its lower melting point (< 1000 °C). Estimates of the loss for ²³⁸U and ²³²Th as given in Table II, are acceptable due to the inhomogenity of NORM in oil sludge waste used.

From the calculation done using RESRAD code, it was possible to predict the maximum total dose received by the workers at the location as given in Table III. ²²⁶Ra was the main contributor compared to other radionuclides and poses a greater hazard due to its longer half-life. Similarly, using the RESRAD code, it was possible to predict the health risk posed to the workers. Estimates show that risk of cancer is low, with ²²⁶Ra again being the radionuclide of concern for the 30 year period as given in Table IV.

Table I: Summary of Specific Activities of NORM radionuclides in oil sludge wastes before and after incineration (Bq/kg).

<table>
<thead>
<tr>
<th>Sample</th>
<th>Specific Activity (Bq/kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>U-238</td>
</tr>
<tr>
<td></td>
<td>Ash</td>
</tr>
<tr>
<td>Range</td>
<td>Before</td>
</tr>
<tr>
<td></td>
<td>17.5 – 37.5</td>
</tr>
<tr>
<td>Average</td>
<td>25.3 ± 6.0</td>
</tr>
<tr>
<td>Total Estimates activity loss (Bq/kg)</td>
<td>133.84</td>
</tr>
<tr>
<td>% change (+/-)</td>
<td>- 4.42</td>
</tr>
</tbody>
</table>
Table II: Summary of Specific Activities of NORM radionuclides in oil sludge wastes before and after incineration (Bq/kg).

<table>
<thead>
<tr>
<th>Sample</th>
<th>Specific Activity (Bq/kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Th-228 (Ra-228)</td>
</tr>
<tr>
<td></td>
<td>Before</td>
</tr>
<tr>
<td>Range</td>
<td>Ash</td>
</tr>
<tr>
<td>12.6 – 18.6</td>
<td>23.7 – 48.2</td>
</tr>
<tr>
<td>Average</td>
<td>15.8 ± 2.3</td>
</tr>
<tr>
<td>Total Estimates activity loss (Bq/kg)</td>
<td>83.58</td>
</tr>
<tr>
<td>% change (+/-)</td>
<td>+ 15.50</td>
</tr>
</tbody>
</table>

Table III. Dose Contribution (in mSv/yr) for All Pathways for every Oil Sludge Radionuclides in 1 to 100 year period.

<table>
<thead>
<tr>
<th></th>
<th>T=1 year</th>
<th>T=3 year</th>
<th>T=10 year</th>
<th>T=30 year</th>
<th>T=100 year</th>
</tr>
</thead>
<tbody>
<tr>
<td>K-40</td>
<td>3.154E-04</td>
<td>2.897E-04</td>
<td>2.150E-04</td>
<td>9.172E-05</td>
<td>4.651E-06</td>
</tr>
<tr>
<td>Ra-226</td>
<td>1.987E-02</td>
<td>1.969E-02</td>
<td>1.906E-02</td>
<td>1.737E-02</td>
<td>1.257E-02</td>
</tr>
<tr>
<td>Th-228</td>
<td>3.449E-04</td>
<td>1.702E-04</td>
<td>1.435E-05</td>
<td>1.224E-08</td>
<td>2.221E-09</td>
</tr>
<tr>
<td>U-238</td>
<td>1.560E-06</td>
<td>1.576E-06</td>
<td>1.633E-06</td>
<td>1.807E-06</td>
<td>2.575E-06</td>
</tr>
<tr>
<td>Total</td>
<td>2.055E-02</td>
<td>2.021E-02</td>
<td>1.948E-02</td>
<td>1.781E-02</td>
<td>1.328E-02</td>
</tr>
</tbody>
</table>

Table IV. Cancer Risk for All Pathways for every Oil Sludge Radionuclides in 1 to 30 year period

<table>
<thead>
<tr>
<th></th>
<th>T=1 year</th>
<th>T=3 year</th>
<th>T=10 year</th>
<th>T=30 year</th>
</tr>
</thead>
<tbody>
<tr>
<td>K-40</td>
<td>4.182E-07</td>
<td>3.840E-07</td>
<td>2.850E-07</td>
<td>1.216E-07</td>
</tr>
<tr>
<td>Ra-226</td>
<td>4.128E-05</td>
<td>4.090E-05</td>
<td>3.959E-05</td>
<td>3.608E-05</td>
</tr>
<tr>
<td>Th-228</td>
<td>8.851E-08</td>
<td>4.366E-08</td>
<td>3.681E-09</td>
<td>3.141E-12</td>
</tr>
<tr>
<td>Th-232</td>
<td>5.231E-07</td>
<td>5.731E-07</td>
<td>7.108E-07</td>
<td>9.413E-07</td>
</tr>
<tr>
<td>U-238</td>
<td>2.872E-09</td>
<td>2.902E-09</td>
<td>3.006E-09</td>
<td>3.327E-09</td>
</tr>
<tr>
<td>Total</td>
<td>4.232E-05</td>
<td>4.190E-05</td>
<td>4.059E-05</td>
<td>3.715E-05</td>
</tr>
</tbody>
</table>

4. Conclusions

There is no significant increase in terms of dose exposure to the workers and the environment. NORM specific activity level in local oil sludge wastes is low and is comparable to normal soil. Incineration process used, has reduced the weight of oil sludge waste by half. From the calculation it has shown that the projected maximum dose exposure to workers from $^{238}\text{U}$, $^{232}\text{Th}$, $^{226}\text{Ra}$ and $^{40}\text{K}$ radionuclides at the disposal site was estimated to be about 2.055 x 10-2 mSv/yr for the first year of deposit of incinerated oil sludge wastes, with a decrease to 1.328 x 10-2 mSv/yr for the 100 year period. These values are still under the annual dose limit for the public and radiation workers as stipulated by the
Malaysian Atomic Energy Licensing Board (AELB). The study has also shown the potential risk to cancer at the site for the first year of about $4.232 \times 10^{-5}$ decreasing to $3.715 \times 10^{-5}$ for a period of 30 years. This suggest that for a one year of activity in disposal of incinerated oil sludge at the disposal site the level projected would not be risky to workers of the industry.

Acknowledgements: We would like to thank supporting staff of the Nuclear Science Programme, School of Applied Physics in assisting this research.

REFERENCES


Abstract. Most people remain unconvinced that living next to a deep repository for radioactive waste (especially high-level waste) is safe. The waste is seen as dangerous, polluting and unpredictable. There is also a feeling in many countries that 'solving' the disposal problem only encourages the nuclear industry to continue its ‘dangerous’ activities. Designing a repository for radioactive waste deviates from standard engineering practice in that no high-level waste (and only a few low- and intermediate-level waste) repositories exist and, even where they do, testing their compliance with design limits is difficult due to the long time scales involved. For most people, such complexities are difficult to grasp, further engendering suspicion and fear. Thus, significant efforts must be expended by those responsible for waste disposal to make it clear to the general public that it is possible to assess and demonstrate the long-term safety of a repository. Such a comprehensive public demonstration effort has not been made in most programmes and in many countries geological waste disposal programmes have encountered fierce resistance, stopping progress towards constructing and operating a repository. Secrecy - or at least the feeling that the experts are not telling the whole truth - compounded by a failure to provide relevant and sufficient information are issues that have always dogged waste disposal programmes. Thus it is important to consider how this information is communicated by experts to non-experts and to evaluate success as a function of progress within any given waste management programme. This paper considers how such communication has been approached in two programmes, namely Switzerland and Japan. As communication ultimately comes down to a local level as part of the siting process, the focus will be on this aspect.

1. The Swiss programme

The Swiss waste management concept currently foresees two types of repository, one for low- and intermediate-level waste (L/ILW) and one for vitrified high-level waste (HLW), long-lived intermediate-level waste (ILW) and spent fuel (SF). Both are deep geological facilities that can be closed definitively and will have the ability to ensure passive long-term safety thereafter. Nagra – the National Co-operative for the Disposal of Radioactive Waste – was established in 1972, with responsibility for all preparatory work associated with implementing disposal projects. Switzerland has a long tradition of formalised, direct involvement of the public in decision-making processes on all political levels. Consequently, over the years, Nagra has built up an open and transparent communication strategy.

On a local level, the situation in Switzerland is probably best demonstrated by the Wellenberg experience. The search for a potential site for a repository for low- and intermediate-level waste was conducted in a scientifically transparent manner. Despite early efforts to communicate with the people at potential sites, some strong opposition was encountered. Following a lengthy narrowing-down procedure, Wellenberg in Canton Nidwalden was finally proposed as the potential site in 1993. Agreements were concluded between the local community and the Canton regarding compensation payments and an intensive information campaign was conducted at a local level. However, in a referendum held in June 1995, the people of the siting Canton (Nidwalden) voted, albeit by a narrow majority, against the general licence and the exploratory drift. Based on an analysis of the voting results and on consultations in a series of working groups, it was felt that modifications to both the disposal concept and the licensing approach would improve the situation. There was an indication of a desire on the part of the people for greater monitoring and retrievability in the concept. It was also felt that asking for both the exploratory drift and the repository licence in one step was too much, the fear
being that if permission for the drift were granted the repository would proceed irrespective of the results from the investigations. A decision was therefore made to apply first for the exploratory drift concession only. Against this background, an application for a concession for the exploratory drift was submitted at the beginning of 2001. However, in September of that year, the people of Nidwalden voted against this application for a second time. Although the siting community continued to be in favour of the project, general opposition in the Canton was actually stronger than it was in 1995.

Despite every effort to communicate and inform openly and objectively, the project was thus blocked on a political level and the site has since been abandoned. An analysis of the reasons for the second failure provides some lessons for future communication efforts. At the time of the second referendum, there was a general crisis of confidence in organisations and politicians on an international scale (Enron, WorldCom, etc.), as well as a general feeling of uncertainty following the events of 11th September 2001. On a more local level, contamination of the traditionally ‘friendly’ atmosphere in Nidwalden had been achieved by a long, emotional and aggressive referendum campaign by the opponents. Faced with this very emotional anti-nuclear campaign, with use of powerful and sometimes disturbing images and a licensing procedure which gave the Canton the power to effectively veto a project, the somewhat dry, factual, unemotional supporters’ campaign was ultimately unsuccessful in dispelling local distrust and concern. Added to this was the fact that, although there was always approval of the project in the siting community, neighbouring communities felt somewhat excluded from the process.

On a national level, two anti-nuclear initiatives (phasing-out of nuclear power and a moratorium on new power plant construction) were overwhelmingly rejected in a referendum held on 18th May 2003. It seems that the public accept the need for nuclear power. Also interesting, particularly in the wake of the Wellenberg experience, is that the new Nuclear Energy Law, in force from 2005, effectively removes the power of the Cantons to veto a repository project.

2. The Japanese programme

The legacy of a series of accidents in Japan (most notably the Tokai bituminisation plant fire in 1997 and the JCO (Japan Nuclear Fuel Conversion Co.) criticality accident in 1999), which were not always handled in the best and most open way by the nuclear industry, has meant that efforts to build the Japanese public’s confidence in waste management have faced a serious challenge. Essentially, by the end of the 1990s, when the geological disposal programme moved from the generic to the implementation phase, a general feeling of mistrust in all things nuclear was present in the Japanese population.

This is illustrated by the Japanese HLW programme. Early decisions in the generic phase of the programme tended to be made by groups of experts under the control of the national government. The tendency was to simply announce the results of discussions, with important decisions, such as the definition of the disposal concept, being suddenly introduced into the public domain as a “fait accompli”. The public had no access to, and no influence on, the decision-making process. With the establishment of the Nuclear Waste Management Organization of Japan (NUMO) in 2000, the HLW programme entered the early stages of site selection. NUMO has opted to use a novel and transparent approach, based on complete reliance on a volunteer process. In December 2002, an information package was sent out to all 3239 municipalities in Japan to provide them with the background needed to consider volunteering for consideration as an area for preliminary investigations. Distribution of the information package has been supported by a wide range of additional activities aimed at making contact with potential volunteers. NUMO held public fora in major cities in 31 of all 47 prefectures in 2001 to 2002; approximately 5,000 participants attended these events. Local media at each location jointly hosted the fora and reported the events as feature articles. NUMO and local newspapers have since jointly hosted round-table talks with local opinion leaders at more than 30 locations, with results reported in the local media as feature articles. In terms of actual siting, it may be difficult to achieve sufficient public understanding and acceptance through the efforts of the implementing organisation alone. The government is therefore expected to clearly define the role of the implementing organization in accordance with national policy. The government is also responsible for setting up a
system for siting, promotion of public understanding of waste policy and proposals for development of local economies and infrastructure.

On a national level, NUMO’s programme has also been advertised in popular magazines and newspapers. A poster campaign was conducted at major train stations in October 2002 and May 2003. Two videos have also been produced to help in understanding the concept of geological disposal. An interactive website to promote public dialogue is also being developed. NUMO strongly believes that public involvement is vital in pursuing a deep geological disposal project and aims to ensure transparency in its work against the background of the requirements of the Final Disposal Act of 2000, which states that the opinions of prefectural governors and city (municipality) mayors must be ‘heard and fully respected’ in the site selection process. A review committee of external experts is also responsible for ensuring information disclosure. NUMO also has two technical advisory committees, one domestic and one international. These ensure technical transparency and open meetings, involving the public, have been held. NUMO also believes that international collaboration plays an important role in contributing to confidence. It is clear that such a volunteer approach can only succeed with the support of local communities and populations. It thus has high international visibility and, if successful, it will represent a significant step forward in opening up the site selection procedure to the general public. If unsuccessful (i.e. no volunteers come forward), there is, however, a risk of a negative reaction to any subsequent nomination of sites. In this sense, public communication in Japan over the next few years will probably be more critical than in any other national programme.

Further information and references on confidence building in these and other programmes can be found in West and McKinley [1].

3. Lessons for the future

Technical experts often forget that waste disposal is an emotional issue and it cannot be addressed simply in a dry technical manner. Feelings run strong (as shown in Switzerland and Japan) and scientists have to learn to address the issues of concern in a more engaging manner. In this sense, a waste disposal programme is like a sporting event such as figure-skating. While technical merit and an ability to perform the technical components of the programme to the required standard are essential, artistic impression is equally important in convincing people to ‘vote’ in favour of a proposal.

Artistic impression is where waste programmes usually lose marks. People cannot appreciate and understand the significance of the technical work and have little or no confidence in it as there is often no positive emotional connection. In terms of building confidence, and the obstacles to be overcome, some of these negative concerns and emotions are generic and can be viewed on an international scale (fear of radiation, mistrust of authorities, the NIMBY (‘Not In My Backyard’) syndrome). However, lessons have to be tailored to each programme’s experiences and culture. The phenomenon of group memory or perception also pays a strong role. People’s perception of a project, and their confidence in it, are necessarily affected by what has gone before and this is quite programme-specific. Project time scales are also an important factor affecting group perception. Waste disposal projects are inherently long-term in nature but this can lead to problems if things ‘drag on’ for too long. For example, after almost eight years of intensive campaigning at Wellenberg, not even an exploratory drift was constructed. People do become tired of an issue and may eventually reach the stage where they simply wish it would go away. It is not surprising, that many programmes score so badly in ‘artistic impression’.

Closely associated with the issue of confidence-building in the nuclear field is the concept of risk. This is probably the most difficult aspect to convey as, in general, many societies are becoming risk-averse, with people wishing to avoid exposure to any perceived threat. This is illustrated by recent health panics involving the outbreak of SARS (Severe Acute Respiratory Syndrome) in the Far East, which caused a severe loss of airline passenger traffic, bird flu and ‘mad cow’ disease, resulting in an ongoing crisis in confidence in food industry. Where health is linked to science, individuals become increasingly worried about risk. Thus, it is important to show that the benefits of a project will outweigh the disadvantages. Unlike other engineering projects such as roads, and even nuclear power plants, there is nothing inherently attractive or beneficial about a waste disposal facility, except
perhaps the compensation payments involved. Relying on people’s wider sense of responsibility – to
the nation as a whole or to future generations – is simply not enough.

Acknowledgement. This paper is published with permission of the Executive Director (BGS).

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Importance of the 'Joint Convention on the Safety of Spent fuel Management and on the Safety of Radioactive Waste Management'

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a Nuclear and Industrial Safety Agency, Ministry of Economy, Trade and Industry, Tokyo,
b Japan Nuclear Energy Safety Organization (JNES), Tokyo, Japan

Abstract. Lessons learned from the preparation of the National Report and the participation to peer review of the Joint Convention are described. The workload for the preparation of the National Report was tremendous but worthwhile because the international review was quite effective for keeping Contracting Parties to the safety level of spent fuel management and of radioactive waste management. It is expected that many more member states will join the Convention in the near future.

1. Introduction

During preparation of the first National Report for the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management (herein after called the “Convention”), the authors participated in the preparation of the draft of the National Report and discussions in the subcommittee on review of National Report for the Convention at Nuclear and Industrial Safety Agency (NISA) of the Ministry of Economy, Trade and Industry (METI) and joined for the First Review Meeting held in November 2003 in Vienna.

The first National Report described the safety regulations and the activities of the operators related to safety of spent fuel management and safety of radioactive waste management in Japan. The completion of the report owed to the co-operation by involved parties over year and a half. By taking this opportunity, we would like to express our gratitude to those who co-operated with us.

The national reports of the Contracting Parties were peer-reviewed internationally at the Review Meeting, and thus they became useful materials that explain the overview of radioactive waste management in Contracting Parties.

The first Review Meeting in November 2003 gave us a good opportunity to become familiar with the information on the outline of regulations of Contracting Parties. As shown in the summary report of the Review Meeting, the next meeting (May 2006) is expected to be a good opportunity for Contracting Parties to exchange information and experiences frankly about legal framework and practices.

Since the framework of joint Convention requires the revision of National Report every three years, the second National Report must be submitted by October 2005. We are now engaged in the preparation of subsequent National Report. Lessens learned from the preparation of the first National Report and the matters to be reflected in the subsequent reports are described in this paper.

2. Comparison of the 'Joint Convention' and the 'Nuclear Safety Convention'

One of the authors was engaged in the preparation of the National Report under the nuclear safety Convention, and commonly featured topics of the respective two National Report are as follows:

- Radioactive wastes are generated as the by-products from the operation of nuclear facilities. The wastes go through treatment, storage and other processes, and finally they are disposed of. The facilities and practices are required to be operated as efficiently as possible.

- To maximize the efficiency of disposal of radioactive waste, each country is also required to make the disposal facilities and methods consistent with other non nuclear industrial waste management and take into consideration non-technical aspects, for example, public acceptance, culture, history and geographical conditions.
- Nuclear power plants, subject to the nuclear safety Convention, are based on the well-established technology. There are “international standards” for technical or legal requirements among Contracting Parties. Meanwhile, radioactive waste subject to the Convention has a variety of categories among Contracting Parties and many of the requirements for the disposal facilities are specific and differ greatly among individual sites and vary greatly. Therefore, it is difficult to define so-called “international standards.” The facilities subject to the Convention is, in a manner, intended for regulations for individual diversified facilities.

- In concrete terms, the radioactive waste from the research reactors, the use of radioisotopes (RI), radioactive waste related to medical care, and those from nuclear power generation and nuclear fuel cycle have a variety of characteristics. Safety management should be optimized depending on these characteristics.

- Requirements for safety management may vary even in the regulatory system of the same country. For example, the spent fuel and radioactive waste that contains a large amount of radioactive materials, the radioactive waste which contains toxic substances and radioactivity, the radioactive waste generated from medical facilities that require consideration for biological influences and the radioactive waste which was generated from small scale use of radioisotopes may be applied to a more appropriate management system.

- To secure the safety of radioactive waste, each Contracting Party should take into account to take measures with a long-term point of view. At this time, it is difficult to develop a regulatory system that covers every situation of radioactive waste management. Therefore, a graded approach and other provisions have been proposed and establishment of a global common framework should be necessary.

3. Matters taken into consideration in preparation of the first national report and matters to be reflected in subsequent reports

Each Contracting Party has an obligation to explain the measures they adopted to meet the requirements (Articles 4 to 28) of the Convention in their national report. However, these requirements are specified to cover the facilities and practices for diversified radioactive waste and spent fuel as described in Section 2. Consequently, some of these requirements are redundant or inapplicable to certain types of facilities.

In preparation of the first National Report, we made an efforts to apply the requirements of the Convention to all the facilities in Japan and to describe what kind of measure is adopted as the highest common factor.

The Convention is incentive in nature and promoting the safety of spent fuel management and the safety of radioactive waste management. From this fundamental point of view, the contents of the report should optimize the descriptions for the facilities to which relatively mitigated application of safety requirements, the facilities that require strict regulatory requirements, such as the facilities of medical radioactive waste, if compared with the facility of high-level radioactive waste.

When emphasis is put on the aspect of information exchanged by the framework of this Convention, the information on the safety of spent fuel management and on the safety of radioactive waste management in Contracting Parties and the information on the situation of implementation of safety regulations on individual facilities must be beneficial to each Contracting Party.

- Hospitals and small-scale facilities Article 2, Definition Clause (f):
  “Nuclear facilities” in the Convention refers to the civil facilities and associated premises, buildings and equipment where radioactive materials are manufactured, fabricated, used, handled, stored or disposed of in a scale that requires consideration for safety.

In the national report, we use “nuclear related facilities” for hospitals and small-scale facilities rather than “nuclear facilities” because hospitals and small-scale facilities are not considered to be subject to the provision of the regulation as “nuclear facility “in Japan.
Requirements for nuclear fuel cycle facilities should be separated from requirements for other facilities. It is obvious that consideration for decay heat is not required in small-scale facilities. Article 11, Clause (i).

- Description of uranium mining waste:

  When the Convention was established, it was promised that the decision as to whether or not uranium mining waste should be reported was left to option of each Contracting Party, depending on whether uranium mining waste is defined as radioactive waste or not. However, it was decided that uranium mining waste management has to be reported in the national report based on discussion at the plenary session of the first Review Meeting.

- Assurance of financial resources not to burden future generations:

  In Japan, financial resources for decommissioning of nuclear power plants, for the reprocessing facilities and for high level radioactive waste disposal have been assured. And the financial resources needed for decommissioning of fuel fabricating facilities and low-level radioactive waste management facilities are covered in the management of the current business operational activities. This is probably because the wastes from these facilities are not so much in volume and are of rather low radioactivity. However, the framework is being established in Japan for the waste generators to bear expense of recycling industrial waste other than radioactive waste. There may be a possibility that some kind of financial measure will be considered for radioactive waste in the future.

- Article 4, specifies “Biological, chemical and other hazards related to spent fuel management shall be taken into consideration.” For the upstream and downstream facilities involved with the normal nuclear cycle, it is not considered necessary to take biological hazards into consideration. However, some RI waste may be generated from facilities such as hospitals. For these, appropriate measures for incineration etc are taken by radioactive waste collecting service agents since there is no regulatory requirement.

- Consideration of technical provision for the decommissioning at the design stage is not stipulated as legal requirement in Japan, but the operators are taking autonomous actions to reduce cobalt contents in materials near the reactor core. This is effective for reduction of radiation level of the workers in the power reactor facilities. It takes a long time period from design stage to decommissioning stage of a nuclear facility, and it is expected that vast progress in decommissioning technology and in research and development will also be promoted in the long time period so that the state of the art technology may be better reflected at the stage of license for the decommissioning plan.

- Environmental impact assessment (Article 4 (iv) and Article 11(iv)).

  In Japan, it is specified that nuclear power plants must implement environmental impact assessment. For other facilities, the necessity of environmental impact assessment is judged on a case by case, depending on the scale of operation and facility.

4. Conclusion

There were some difficulties during preparation of the national report of Japan, but through the process, we are aware of a lot of important and basic issues related to the safety of spent fuel management and the safety of radioactive waste management. These issues were some lack of consistency to international standards and gaps between ideal situations and actual situations which reflect history of our nuclear communities. However, we believe, that the Convention is quite effective to promote the safety of management of spent fuel and the safety of radioactive waste management of Contracting Parties.
LILW repository in Slovenia: uncertainty treatment in performance assessment of Slovenian generic safety case

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Abstract. The Slovenian Agency for Radwaste Management has founded and developed a PA team, which has been educated also through the scholarship and other help of the IAEA, to perform performance assessment calculations for Slovene LILW repository safety case. The team has gone through several steps of the performance assessment procedure, starting with scenario generation and ending with radionuclide transport calculations for each of selected scenarios. To do this, two Slovene sites have been chosen and preliminary designs of a near surface and an underground repository have been made. In 2004, the team was asked to concentrate on the part of confidence building process concerning the treatment of uncertainty for the post closure assessment. The team has thus treated all kinds of uncertainty in PA/SA: scenario uncertainty, model (conceptual model, mathematical model, computer code) uncertainty, data/parameter uncertainty and subjective uncertainty, each of these groups in a specific way, recognised by the international practice. We used deterministic calculations to address scenario and model uncertainties. Scenarios, which have thus been calculated, were: normal evolution scenario with progressive engineered barrier system degradation, cap failure scenario, instant failure of all engineered barriers, climate-change scenario, inadvertent human intrusion scenario - all of them assuming different paths through the biosphere. Model uncertainties have also been addressed by deterministic calculations: calculations of different conceptual models and comparison of results, combining numerical solutions with theoretical ones and using different computer codes for the same conceptual models. Data/parameter uncertainties have been treated through sensitivity analysis (single parameter variation – deterministic) and multi-parameter variation (stochastic calculations), using all available site-specific data. Subjective uncertainty we have addressed through thorough international revisions of the accomplished work. Throughout the whole work, quality assurance procedures, which had previously been developed for the sake of PA/SA calculations, have been kept.

1. Introduction

Slovenian PA/SA team has been founded in 1998 by Slovene Agency for Radwaste Management. The team leader took part in the IAEA co-ordinated 3- year international ISAM project (Improvement of Safety assessment Methodologies for Near-Surface Radioactive Waste Disposal Facilities), where internationally accepted methodology for PA/SA has been set up. The team started with scenario generation work, and then continued under the guidance of Belgian SCK/CEN experts with the work on conceptual and mathematical model generation. The methodological set-up for calculations has been made and first applications of computer codes on radionuclide transport calculations at generic (imaginary) site in Slovenia have been performed. Through years, the methodology and calculation techniques have been developed for more complex safety cases and the last step in 2004 was devoted to the treatment of uncertainty through calculations.

2. Deterministic calculations

2.1. Description

Deterministic calculations (calculations with previously set models and predetermined parameter values) we have run by computation of radionuclide transport on two locations of a site and for two different types of repository: surface (vault type) located on gravel and underground repository, some
70 m beneath the surface, in a silt soil with lower water permeability. Inventory consisted of 18,000 m³ of LIL-waste mainly from the only NPP in Slovenia. The waste was solid – no liquid or gas phase is accepted, with mainly short–lived and only a few long-lived isotopes, like: Ni-59, Nb-94, I-129 and others. Besides, two decay chains have been considered: uranium and plutonium.

Most of waste is packed into 864 l tube type containers, some in 200 or 320 l drums. In the surface repository, the drums are sealed in concrete and disposed in 10 vaults, 20x20x10 m. After closure, the vaults are topped with low permeability capillary cap. Water from repository is drained.

In the underground repository, the engineered barriers are nearly the same as for the surface type, except that instead of vaults, the waste is disposed into 16 disposal tunnels, sealed with concrete. The tunnel lining structure represents also a component of the multi-barrier system.

Calculations have been run first for the radionuclide transport through the near field, then followed by transport calculations for the geosphere and finally for the transport through the biosphere. Computer codes used were: for the near field: PORFLOW and HYDRUS, for the geosphere GMS and for the biosphere AMBER, RESRAD and BIOPHARE. Results have been given in a form of the annual effective equivalent dose to a member the critical group, compared to the Slovene and international dose constraints. The assumptions used have primarily been cautious and the conservativeness supported either by different expert opinions or by parallel calculations. Through the assessments, it has been supposed that environmental and societal conditions on the site would not change and that design, construction and monitoring would avoid major defects. The time of active institutional control has been taken for 100 years and the passive another 200 years. Calculations have not finished till the peak values of radionuclide concentrations in water have been reached.

2.2. Calculated examples and results: scenario, conceptual model, mathematical model, computer code, data–parameter calculations

Deterministic calculations have bee used to fight the scenario uncertainty for different future evolutions of the site and for some cases of disruptive events. Thus, we have calculated radionuclide transport for the normal evolution scenario - with progressive engineered barrier system degradation, cap failure scenario, instant failure of all engineered barriers (road construction scenario), climate-change scenario - all of them assuming different pathways through the biosphere and inadvertent human intrusion scenario (direct intrusion for surface repository and well drilling for the underground). The highest doses came out from the road construction scenario. They were the only doses to exceed the present dose constraints in Slovenia.

To fight the model uncertainty: conceptual model, mathematical model and computer code uncertainties, numerous deterministic calculations have been performed. Under conceptual model uncertainty, we studied effects of different radionuclide flow paths through the geosphere and through the biosphere, calculating doses for two different repository cap types, effects of different thickness of the unsaturated zone beneath the surface repository, and two different types of the backfill (sand and bentonite) for both repository types. We also studied effects of different waste distributions within repository and showed that the most favourable results come, if the more active wastes lie in its upper part.

For mathematical model uncertainty studies, we varied the grid density – for the near field and for the geosphere calculations, comparing results form two 3D to 2D and finally to 1D calculations to prove the conservative of results.

Computer code uncertainty we addressed by calculating numerical dispersion for transport through the geosphere. Besides, we performed parallel calculations for the radionuclide transport through the near field and through the biosphere using different computer codes: PRFLOW/HYDRUS and AMBER/RESRAD/BIOPHARE. For the geosphere calculations, comparison of calculated results with analytical solution and experimental data has been made (slug injection test) and showed good agreement of results.

Results: through conceptual model calculations, we optimised the designed parameters of the surface repository. We have also shown the conservative approach from 3D to 1D calculations and we have shown that computer codes used, give realistic results.
Deterministic calculations for parameter uncertainty treatment have been used in numerous cases: many of such calculations have been performed through the calibration of hydro geological model of the site (area of 82 km²). We finally succeeded in getting the 3D model, calibrated with measured data from the site. The same situation had to be solved when setting the near field model of repository (surface and underground) and with 3D to 1D calculations, with necessary subsequent parameter adjustments.

3. Probabilistic/stochastic approach

Data/parameter uncertainty has also been treated systematically by stochastic approach. We used single parameter variation method for the sensitivity analysis on parameters in the near field and biosphere (dose sensitivity) calculations.

8 parameters (for concrete and soil) have been treated in the uncertainty analysis for the near field, each of them being characterized by its own probability density function. These parameters were: Distribution coefficient \(k_d\), molecular diffusion, hydrodynamic dispersion, solid density, effective porosity, total porosity, and dilution factor and biosphere conversion factor. By calculations, four radionuclides, with the highest transport fluxes from the near field have been treated: H-3, Ni-59, I-129, and Tc-99. Parameter values, together with their distributions have been used as entering values in Monte Carlo calculations. These calculations have been performed by our home-developed code, the idea of which was based on the SCK-CEN LISA computer code: after having built the 1D transport model for PORFLOW, the PORFLOW input data have been supplemented by parameters of uncertainty for each material type. The outputs were: correlation among parameters, correlation between parameters and the dose and a set of fluxes over time.

To determine which of the site-specific parameters were critical for the values of biosphere conversion factor (BCF), we varied these parameters and observed the effects of variation on the BCF. Each parameter has been varied within \(\pm 50\%\) of its base value. The resulting distribution was characterized with statistical indicators and the results sorted by them. Through this procedure we have identified the parameters, which have the greatest influence on the BCF and thus need to be high on the priority list when gathering site-specific data.

Results: The most significant factor for the transport of most of radionuclides was the distribution coefficient (not the case for tritium, which is not retarded).

For the dose sensitivity calculations, we have performed variation over human, plants and cattle input parameters and seen the effect on the BCF. The parameters have been varied by multiplying the base value with a randomly generated value that was chosen from a uniform distribution from 0.5 to 1.5, effectively varying each parameter randomly between 50% and 150% of its base value. One thousand random runs have been performed for each of the varied parameters.

Result: the largest contribution for all radionuclides is from grain crops pathway. Similarly, the concentration of radionuclides in crops depends heavily on the interception factor (from contaminated water) and the depth to which the contaminated water soaks into the ground. Variations in other parameters by \(\pm 50\%\) produce at most 3% standard deviation, so we can conclude that the BCF was not particularly sensitive to them.

4. Conclusion

After three years of radionuclide transport calculations from surface and underground repository in Slovenia and its consequences on the critical group, the Slovenian team performed systematic uncertainty analysis on results obtained. We have treated scenario uncertainty, conceptual model, mathematical model and computer code uncertainty through deterministic calculations, while data/parameter uncertainty has been treated in two different ways: by deterministic calculations, and by stochastic calculations – performing sensitivity analysis through single parameter variation method. Calculated results, together with developed quality assurance procedures and international revisions have shown that performed assessments have been put on the firm ground so that we could start our work on the real site. The site selection procedure for repository is today the first priority for Slovene
Agency for Radwaste management and firs assessment calculations are expected to start already in this year.

REFERENCES


Establishing of repository for low and intermediate level waste in Denmark

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Abstract. The Danish Parliament has in March 2003 agreed to start the work to establish a ‘Basis for Decision’ on a Danish disposal facility for low and intermediate level waste. The radioactive waste comprises waste from the operation of three research reactors and other nuclear facilities at Risø National Laboratory as well as radioactive waste from use of radioactive materials for medical, industrial and research purposes in Denmark. By far the largest part of the Danish waste is LILW-SL. However, a small amount is considered to be LILW-LL. Three repository types could be considered: 1) a near-surface repository, potentially combined with a deep borehole for the small amount of long-lived waste, 2) a deeper near surface repository located maybe 30-80 m below the surface, and 3) a geological repository. If the ‘Basis for Decision’ is accepted, the next step will most likely be to make a geological survey and a feasibility study of the potential repository types. Based on this a few sites should be selected for further investigation.

1. Introduction
The Danish Parliament has in March 2003 agreed to the costs and the general decommissioning approach for all the nuclear research facilities at Risø National Laboratories. The objective is to decommission all nuclear facilities at Risø as soon as possible within a timeframe of 20 years. At the same time, the Parliament agreed to start the work to establish a ‘Basis for Decision’ on a Danish disposal facility for low and intermediate level waste (LILW).

The ‘Basis for Decision’ on a final repository is scheduled to be presented to the Danish Parliament in the fall 2005, and must describe how to carry out the project of establishing a final repository. Both the issues related to the decision approach, the technical issues and the public information issues must be described. These issues include e.g. safety and environmental principles, waste description, possible repository type as well as how to inform the public and ensure transparency.

2. Waste in Denmark
The overall policy and practice for radioactive waste management have so far been to collect and store all Danish radioactive waste under safe and secure conditions at dedicated storage facilities at Risø. The stored radioactive waste comprises waste from the operation of the three research reactors and other nuclear facilities at Risø as well as radioactive waste from use of radioactive materials for medical, industrial and research purposes in Denmark.

The latest inventory of waste and waste types are presented in the Table 1 below. The decommissioning waste is an estimated amount.

By far the largest part of the Danish waste is LILW-SL. However, a small amount is considered to be LILW-LL. Denmark does not have any HLW.

The final repository is expected to accommodate at least 5.000 m³ conditioned waste.
Table 1: Latest inventory up-date of Danish waste as of 1.1.2005

<table>
<thead>
<tr>
<th>Weight /volume</th>
<th>TBq activity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Irradiated uranium 233 kg</td>
<td>0.23 t</td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td>LILW (conditioned)</td>
<td>1200 m$^3$</td>
</tr>
<tr>
<td>LILW (not conditioned)</td>
<td>125 t</td>
</tr>
<tr>
<td>Waste (tailings) and malm** from uranium exraction tests</td>
<td>4800 t</td>
</tr>
<tr>
<td>Waste from decommissioning</td>
<td>1000 t</td>
</tr>
<tr>
<td>Tritiated water</td>
<td>16 t</td>
</tr>
</tbody>
</table>

* 18 TBq long-lived $\beta/\gamma$-and 4 TBq $\alpha$-emitters
** Malm is not considered waste

3. Repository type

The type of repository is at the present moment not decided, and a long and democratic process is to be carried out before the decision is finally taken. The final choice will among others depend on adequate geological formations, precise amount of long-lived waste as well as the question of reversibility.

Regarding geology, Denmark has sedimentary geology of primarily sand, clay and limestone. Furthermore, the water table is generally located relatively close to the surface, which can make it difficult to establish a dry near-surface repository.

Three types of repository could be considered:

1) a near-surface repository. Potentially combined with a deep borehole for the small amount of long-lived waste. The near-surface repository can either be above or below the water table.
2) a deeper near surface repository located maybe 30-80 m below the surface.
3) a geological repository.

4. The decision process

At the moment, the “Basis for Decision” for the establishing of the repository is being prepared and it should be ready by fall 2005. This “Basis for Decision” is especially focusing at the fundamental principles for safety and environment. These principles are to set the frames for the rest of the process and will be based on the nine principles of IAEA [1]. There will also be proposed principles for dose constrains for the repository in the “Basis for Decision”. Furthermore, there will be a description/discussion of which technical issues to look upon (e.g. reversibility, waste container).

It is the intention to make the process as transparent as possible to the public thereby allowing stakeholders to participate actively in the decision process. A leaflet has been prepared with information on the project. The leaflet was sent to all municipalities and a number of interested parties and NGO throughout the country. The leaflet is downloadable from the homepage of the Ministry for the Interior and Health.
Following this, it is the intention to have one to two mini-seminars for interested parties. All working papers will be accessible at the homepage of the Ministry for the Interior and Health, in order to allow interested parties to follow the process.

5. The future process

If the “Basis for Decision” is accepted by the Parliament, the next step will most likely be to make a geological survey and a feasibility study of the potential repository types. Based on this, a few sites should be selected for further investigation.

REFERENCE

Safety management of radioactive waste at the centralized facility in Albania

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Abstract. Since 1999 a new radioactive waste management centralized facility is constructed in Albania for the processing and the interim storage of radioactive wastes in accordance with internationally accepted criteria. The safety assessment of this facility was performed considering its impact in the public and the environment. A national legal framework was established already in Albania, providing among others an institutional control in all steps of the waste management procedure. The design of the low activity waste storage is done based in the IAEA recommendations, as well in the daily practice taking into consideration the country particularity. The contemporary methods for the safety procedures are under operation at the centralized facility for waste pre-treatment, processing, conditioning, transport and interim storage.

1. Introduction

The radioactive waste has been generated in Albania during forty years, mainly low level activity waste, which arisen from the use of radioactive materials in medicine, industry and agriculture, from the technological processing on various minerals containing natural radionuclides and from various research and education activities.

The radionuclides of the waste comprise: $^{226}$Ra, $^{241}$Am, $^{137}$Cs, $^{90}$Sr and $^{60}$Co as spent sources, and $^{65}$Zn, $^{90}$Sr, $^{60}$Co $^{99m}$Tc, $^{131}$I, $^{32}$P, $^{57}$Co and $^{63}$Ni, as contaminated objects (clothing, papers, glassware and other materials). The radioactive waste presents approximately some tenth cubic meters (conditioned volume) generated in hospitals, industry, research and education.

The Radiation Protection Act has been approved by the Albanian Parliament in 1995, and based in this Act the Radiation Protection Commission (RPC), as Regulatory Authority in the country was established [1]. During its activity RPC has approved a set of regulations, related with law requirements in the field of radiation protection and safety: licensing, authorisation, inspection and enforcement. Actually are approved the following regulations; licensing and inspection of activities with radiation sources, safe use of radiation sources, safe transport of radioactive materials, safe management of radioactive waste. These regulations have been approved after consultations with IAEA experts.

During the site selection, design and construction of centralized facility, the commissioning of this facility and applications of working procedures we have taken into account relevant features that might affect the safety of the facility.

Based on the regulations for the safe management of radioactive waste the Institute of Nuclear Physics (INP) is in charge for the management of radioactive waste on national level [2].

2. Safety aspects of radioactive waste management

The activities related with radioactive waste management are based in the principles of radiation safety as; the protection of workers, public and environment by radiation detriment, the volume reduction and full confinement of the waste, as well the dose rate in their surface to be in accordance with accepted standards [3].

Regulations on the safe management of radioactive waste classify the waste into two broad categories, based on the half-life of radionuclides: (a) radioactive waste with a half-life of less than 60 days, and (b) radioactive waste with a half-life of more than 60 days.
Liquid radioactive waste is collected in special containers, which after filling up are transported to the Institute of Nuclear Physics for further processing. Solid radioactive waste is collected in special containers and after their filling up, is transported to the Institute of Nuclear Physics for conditioning and interim storage [2]. All activities of the centralized facility are described in the operational manual, which covers all relevant safety aspects of the procedures, in particular:

- Waste receipts (checking of waste in arrival, measurements, assessment of compliance with acceptance criteria, documentation of waste accepted into facility);
- Waste handling, processing, and emplacement;
- Radiation protection procedures and monitoring;
- Limits and conditions for the release of control;
- Monitoring and inspection of the integrity of packaged waste in order to identify possible deterioration;
- Responses to anticipated operational problems, incidents and accidents;
- Safeguard measures;
- Documentation required and procedures.

The special measures are taken to prevent as much as possible the radiation exposures of the workers, which may arise, from external irradiation and from the incorporation of the radionuclides. In order to comply with worker dose limits and also with ALARA principle, protection provision have been taken account of in the design process of the facility.

In countries where the movement of waste to a centralized facility is practiced, special attention needs to be paid to transportation. The transportation of spent radiation sources to the supplier, as well as the waste packages to the site for further processing at the centralized facility in INP, were carried out in accordance with requirements set forth by national and internationally regulations for transport of radioactive materials. Important operational parameters for the waste processing have been defined already, in order to assure compliance with transport criteria. The maximum dose rate in the surface of the waste package has to be below 2 mSv/h. The maximum level of contamination outside of the waste package has been fixed at 4 Bq/cm² for beta and gamma radionuclides and 0.4 Bq/cm² for alpha radionuclides.

Albania has actually small waste quantities, originated from different practices during four last decades, and perhaps it is inappropriate to construct and operate a national repository for disposal of radioactive waste. Safe disposal of radioactive waste, with low and intermediary level activity, is an important question for the country. Our estimation, referring the development of nuclear methods and activities in the country, show that the capacity in storage area is at least sufficient for the next 30 years in Albania. From the feasibility studies it is seen that the near surface disposal facility for low and intermediary radioactive waste fits better for further activities on the future in the field of peaceful nuclear application to our country.

3. Waste management facilities in Albania

The waste management facility should be equipped with the best instruments available. The old interim storage facility (5.0 m x 4.0 m x 3.0 m), constructed outside INP territory since 1970, has been a simple premise on the ground surface with concrete and brick construction.

Since 1999 started the functioning of new centralized waste management facility (17.00 m x 16.00 m x 3.20 m) in accordance with internationally accepted criteria, which is situated within the INP territory. The facility consisted by two main parts: operational area and interim storage area. Apart from the office and sanitary space the building contains the following areas:

- Area for preparing the cement mixture,
- Waste receipt area for checking delivered waste and their documentation,
- Operating area for conditioning the wastes,
- Decay-storage area for waste with short half-lives,
Operational storage area for the interim storage of conditioned wastes (300 m³).

Decontamination and decommissioning (D&D) processes of the old interim storage facility began in 1999 and for that reason the staff of RAW management laboratory transferred at the new facility all plastic bags, lead containers, 200 litter drums, conditioned with spent radiation sources and solid and liquid (vessels) with radioactive waste [5].

For the moment in the interim storage facility is thirteen conditioned drums, where the total activity is about 222 GBq and there are three drums in the conditioning process. Radioactive waste with or without shielding was successively placed into drum, until an activity of about 20 GBq has been reached.

Apart from the area used for preparing the cement mixture all other areas are part of the controlled area of the building. At the access points of the facility for the personnel the equipment for contamination monitoring are installed. The building represents a solid concrete construction. All entries to the operated, storage and disposal areas of building are protected with security locks, PIN code, magnetic switch, as well they are equipped with alarm systems. There is a fire brigade near the INP, trained on radiation protection and safety.

4. The equipment used at the centralized facility

Institute of Nuclear Physics has participated in some national and regional IAEA projects. Equipments, training and expertise have been received through these projects. The facility has been well equipped through the TC project ALB/4/008 “Improvement of Radioactive Waste Management” (2002 – 2004). A concrete drum mixer machine (AT-208) with capacity 100 litter is used to mix cement, sand and water. The rate of cement, sand and water is applied based on IAEA recommendations after their weighing in industrial balance (Adam-150 kg). After mixing by electric vibrator type AT-231E, the test of the concrete physical properties is performed before filling the drum. The test is carried out by hand operated machine type KD-KL-150M, compressing the cylindrical sample with diameter 15 cm x 30 cm by 300 -1500 kN force.

The volume reduction of the solid radioactive waste is carried out using the compactor type SC-55. Air filtration system is in operation during compaction process. A ventilation system with Hepa filter is in operation during the conditioning process of radioactive waste at the operating area. A manual mobile lift device (the maximum loading weight 1200 kg) is used to transfer the conditioned drums, to interim storage area [6]. The dose rate on surface drum and surface contamination are done using the portable radiation monitor PM1402M. For the characterization of the conditioned spent radiation sources are used beta and gamma detectors BD-04 / BD-05. The hand held equipments (Field-Spec – TARGA) for measurement of neutron and gamma radiation are used too.

5. Conclusions

- The radioactive waste management is an important issue. The Radiation Protection Commission has paid special attention to this question and has approved the regulation and the code, in accordance with international accepted criteria.
- For short-lived radionuclides radioactive waste, their disposal is carried out, after sufficient storage, in accordance with half-life and subsequent discharge as ordinary waste.
- The conditioning and interim storage is foreseen for radioactive waste with long-lived radionuclides. The waste generators provide the financial support for these activities.
- The safety aspects of radioactive waste management are considered as an important problem, which are treated in accordance with IAEA Basic Safety Standards.
- Disposal of radioactive waste for short-lived radionuclides is solved as mentioned in this paper. For long-lived radionuclides the possibility of design for near surface disposal facility is under review.
REFERENCES


Preliminary safety assessment for a radioactive waste disposal facility in Cuba.

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Abstract. In the framework of the Swedish-Cuban Collaboration Project, in the thematic of Methods for Environmental Impact Assessment (EIA), the Center for Radiation Protection and Hygiene (CPHR) with the participation of Sweden specialists from Kemakta Konsult AB developed a safety case, in order to evaluate the preliminary radiological impact from a Borehole concept designed to disposal the spent sources from nuclear applications. Details about the adopted methodology, the selected scenarios, applied models and main assumptions are given. The final results showed at preliminary stage, the different radionuclide influence in the calculated doses and the $^{241}$Am and $^{226}$Ra were identified as the most relevant. For the evaluated scenarios the radiological impacts for the disposal facility are below the adopted regulatory limit (0.1 mSv/y) for the critical group. This study supports the new researches in Cuba to give solution to radioactive waste disposal.

1. Introduction

The CPHR is responsible for developing and implementing a national strategy for the safety radioactive wastes management. In Cuba the wastes arrive from the nuclear applications, the hospitals, the industry and the research institutions are the main sources. The Cuban legislation defines obligatory rules in order to provide a safety radioactive waste management including the final disposal to prevent the possible negative impact in the human health and environment. The common solution is the waste isolation into the disposal facilities, according the radioactive waste levels the facility design ranges from near surface facilities for low and intermediate wastes to geological disposal for high level waste. The spent sources are an important problem in several countries due the presence of long-lived isotopes in the inventory and therefore the timeframe of concern are bigger than defined acceptance citeria for near surface disposal facility. The borehole concept is a possible solution for this kind of wastes and could be an acceptable alternative for developing countries. The safety assessment (SA) for radioactive waste disposal facilities is an important tool to demonstrate that a disposal practice is acceptable or not showing compliance with the regulatory criteria, the SA can be used in several activities, for example, in the site selection process, the definition of the waste acceptance criteria, designing and optimising of engineered barriers. In this paper in the framework of collaboration Project [1] was adopted the SA methodology developed in the International Atomic Agency Programme “ISAM” [2]. The SA process includes several steps; but only some methodological details were included in this paper.

2. The safety assessment

The borehole disposal concept (BDC), is a facility designed to disposal the spent sources, this conceptual design is similar to adopted in previous studies [3,4] and the BDC should minimise the need of maintenance and should be supported for the favourable characteristics of the site, in order to reduce the environmental and human impact. From the site selection process in the Central part of the country [5] was identified a region in granodioritic rock, with favourable conditions where was evaluated the BDC.

The BDC includes a vertical borehole drilled in surface (100 m deep), about 300 mm thick, in the weathered zone, the well is protected by a PVC casing to keep loose material from falling in. At the top of well, a bottom plug is provided to ensure the disposal volume keep dry during the operational
period. The disposal containers for spent sources are metal drums lined in concrete to give the necessary protection. The safety assessment goal was to evaluate the radiological impact from the spent sources disposal in a reference disposal facility. According the wastes characteristics an assessment timeframe of $1 \times 10^6$ years was adopted; the selected endpoints were the annual individual doses and the dose constraint for public member 0.1 mSv/y was adopted.

2.1. The borehole site

The BDC needs a site with favourable characteristics, stable zone, poor hydrology conditions, etc. Therefore this concept was evaluated in a geologic site; this was selected taking into account the international requirements for radioactive waste disposal [5]. The lithology is associated to igneous rock (granodiorite), the site is placed in tectonic stable block and the area has very poor hydro-geologic conditions. Other social-economical features of the site are: the main economic activities are cattle, small agriculture practices and the presence of disperse population.

2.2. The radioactive wastes inventory

Only spent sources will be dispose in this BDC, according the relevant inventory for the safety assessment 4 types of sources were selected, see table 1.

<table>
<thead>
<tr>
<th>Radionuclides</th>
<th>Inventory Bq</th>
<th>Application</th>
</tr>
</thead>
<tbody>
<tr>
<td>Co$^{60}$</td>
<td>$6.6 \times 10^{15}$</td>
<td>Industrial Radiography</td>
</tr>
<tr>
<td>Cs$^{137}$</td>
<td>$1.27 \times 10^{13}$</td>
<td>Medicine</td>
</tr>
<tr>
<td>Ra$^{226}$</td>
<td>$1.33 \times 10^{16}$</td>
<td>Medicine</td>
</tr>
<tr>
<td>Am$^{241}$</td>
<td>$6.6 \times 10^{15}$</td>
<td>Smoke detectors</td>
</tr>
</tbody>
</table>

2.3. Scenarios selection

According the characteristics of the disposal system and the BDC and applying the ISAM methodology, two scenarios were selected. The first one assumes, the release of radioactive contaminants from the Repository, the transport through the geosphere and finally the radionuclides arrives to a well, where is intake for the man by water ingestion. The second scenario evaluates the impact by the consumption of cow products (meat and milk) contaminated.

2.4. The model

The applied software was the Model Maker version 4 [6], to create models using the compartments approach, in order to simulate the behaviour of the disposal system. The adopted conceptual model for the BCD, see Fig. 1, took into account the system description and the defined scenarios. The model includes 7 compartments for the near field, 3 compartments for the waste form and the disposal containers, the other compartments represent the backfill and the concrete walls. This approach

![FIG. 1: Conceptual model adopted for the borehole safety assessment](image-url)
allowed modelling the process associate to the release and transport of radioactive material and its relationship with the barriers in the BDC. The dominant transfer processes included in the model were; the dispersion and advection of radionuclides between the different compartments.

The geosphere was divided in 10 compartments and the radionuclide transport by advection and dispersion were the main process modelled. Finally, all radioactive contaminants arrive to a well, where the man and the cows, via water ingestion intake the radionuclides. Finally, the doses for the critical group were calculated, some relevant dose factors [7] are listed in the Table 2.

Table 2: Dose conversion and transfer factors for the evaluated scenarios

<table>
<thead>
<tr>
<th>Radionuclides</th>
<th>Half life years</th>
<th>DC ing Sv/Bq</th>
<th>Tfactor milk d/l</th>
<th>Tfactor meat d/Kg</th>
</tr>
</thead>
<tbody>
<tr>
<td>Co $^{60}$</td>
<td>5.27</td>
<td>3.5 E-09</td>
<td>3.0 E-04</td>
<td>1.0 E-02</td>
</tr>
<tr>
<td>Cs $^{137}$</td>
<td>30.0</td>
<td>1.3 E-08</td>
<td>7.9 E-03</td>
<td>5.0 E-02</td>
</tr>
<tr>
<td>Ra $^{226}$</td>
<td>1600</td>
<td>2.8 E-07</td>
<td>1.3 E-03</td>
<td>9.0 E-04</td>
</tr>
<tr>
<td>Am $^{241}$</td>
<td>432</td>
<td>2.0 E-07</td>
<td>1.5 E-06</td>
<td>4.0 E-05</td>
</tr>
</tbody>
</table>

2.5. The results

The results are summarized in the table 3 and the figures 2 and 3, the dose for meat and milk ingestion are included in the cow product consumption scenario. The higher doses are associated to the water ingestion scenarios, similar to the Borehole safety case developed by ISAM [4] where, the importance of the water pathway was defined. The more important radionuclides in both scenarios are the $^{241}$Am and $^{226}$Ra, this is a logical results, taking into account the associated inventory and half life. The times of the peaks are affected for the geosphere characteristics, the granodiorite rock is a favourable geology to waste disposal since its poor hydrogeology conditions and retardation characteristics increase the radionuclides transport time.

Table 3: Dose peaks and occurrence time for each radionuclide and scenario.

<table>
<thead>
<tr>
<th>Radionuclides</th>
<th>Water ingestion scenario</th>
<th>Cow products consumption scenario</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dose peak mSv/y</td>
<td>Time years</td>
<td>Dose peak mSv/y</td>
</tr>
<tr>
<td>$^{60}$Co</td>
<td>3.45E-08</td>
<td>200</td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>6.10E-09</td>
<td>400</td>
</tr>
<tr>
<td>$^{226}$Ra</td>
<td>8.81E-06</td>
<td>52000</td>
</tr>
<tr>
<td>$^{241}$Am</td>
<td>9.80E-05</td>
<td>900</td>
</tr>
</tbody>
</table>

FIG. 2: Annual dose Sv/y. Water ingestion scenario
The calculated dose is always below the dose constraint limit (0.1 mSv/y) for the public member. Preliminary, according to evaluated assumptions (assessment context, model, scenarios, etc), the Borehole concept is acceptable for the disposal of spent sources. These results are very preliminary results, because the evaluated conceptual design is under developing, and new changes in the design are expected in the future. Therefore, the new improvements to the BDC must be followed.

3. Conclusions and recommendations

It is concluded that the main radiological impacts, for the evaluated scenarios, are associated with the doses by water ingestion and are always below the dose constraint for members of the public. The dominant radionuclides are $^{241}$Am and $^{226}$Ra for both scenarios.

It is recommended to continue developing the SA BDC safety case, to improve the models and to include other important scenarios (agriculture practices, human and biological intrusion, barrier performance, etc).

REFERENCES

Safety and reliability of Rokkasho LLW disposal center

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Abstract. Since 1992, at Rokkasho LLW Disposal Center, low-level wastes (LLW) from nuclear power plants in Japan have been disposed of. The disposed amount of LLW has reached approximately 174,000 drums. In order to dispose of LLW safely and steadily, it is essential to assure quality of disposal works and to assess safety of facility in advance.

1. Introduction
The Rokkasho LLW Disposal Center, located in Aomori-Ken, which is the northernmost prefecture of the Mainland of Japan, is the only operating repository for radioactive waste in Japan, and its disposal target is limited to LLW from operating nuclear power plants. The Center has been operated since December 1992, and now, two Disposal Areas are available for its disposal. No.1 Disposal Area is designed for disposal of homogeneously solidified waste, such as 200 litre drums with solidified liquid waste with cement. No.2 Disposal Area, which has been operated since October 2000, is designed for disposal of solidified dry active waste, including metallic piping, plastic material, etc.

As of the end of March 2005, the No.1 and No.2 Disposal Areas have each disposed of approximately 136,000 and 39,000 waste drums, respectively. Fig. 1 shows past disposal amount and trend of Rokkasho LLW Disposal Center.

2. Outline of Rokkasho LLW Disposal Center
The Rokkasho LLW Disposal Center is composed of two disposal areas, and affiliated facilities, which are mainly installed in LLW temporary storage and inspection building ("LLW Inspection Building"). In this section, the disposal areas and the LLW inspection building are described.

2.1. Disposal Areas
Fig. 2 shows a photograph of the disposal areas. The right hand side is the No.1 Disposal Area, and the left hand side is the No.2 Disposal Area. Each disposal area has repositories made of reinforced concrete, and each repository is called as concrete pit.

Inside of a concrete pit, the porous concrete layers that have a function to collect water are placed. The collected water is drained to outside of the concrete pit. After emplacing the waste drums in a concrete pit, space in the concrete pit is filled with mortar. And then, the top of a concrete pit is covered with a reinforced concrete lid. In order to reduce an impact of groundwater in the future, surroundings of concrete pits is to be covered with bentonite/sand mixture. Moreover, the top of bentonite/sand mixture is to be covered with neighbouring soil.

In the No.1 Disposal Area, 30 concrete pits have been constructed, and each concrete pit can contain approximately 5 000 waste drums. In the No.2 Disposal Area, 4 concrete pits have been constructed, and 4 concrete pits have been under construction. Each concrete pit can contain approximately 13 000 waste drums. In the future, we will extend the number of concrete pits to be able to dispose of approximately 200 000 waste drums at each Disposal Area.
2.2. LLW Inspection Building

LLW Inspection Building is mainly separated into the temporary storage area and the inspection area. The temporary storage area can contain 3200 waste drums. In this area, two overhead traveling cranes to handle transport containers for waste drums are installed. In the inspection area, waste inspection equipment is installed. Operators can check waste drums in the central control room, by observing visual information using that equipment. After finishing the inspection of waste drums, they are placed in horizontal position. Then, 8 waste drums in a row are loaded in a transport vehicle by a special overhead travelling crane, and then, transported to Disposal Area.
3. Quality assurance for LLW disposal works

In order to accomplish safe and reliable operation, JNFL has recently enforced quality assurance activities concerning disposal works. The contents of the quality assurance activities are provided in Safety Preservation Rules of JNFL, and a regulatory body inspects the situation on the observance of the rules quarterly. In this section, we shall describe the outline of Quality Management Systems, and Quality Assurance Activities related to waste drums and disposal facilities.

3.1. Outline of quality management systems

In Rokkasho LLW Disposal Center, General Standards for Quality Management ("the General Standards") are provided as fundamentals for establishment and maintenance of Quality Management Systems. The General Standards is made to satisfy the needs of customers and the public, and to enable activities for disposal business to run well. The General Standards are applied to planning, design, construction, operation and maintenance related to disposal facilities, and also to activities to satisfy requirements related to disposal business as well.

3.2. Quality assurance for waste drums

Japanese regulations provide the technical standards for waste drums. The technical standards consist of specifications for drums and solidification materials, radioactivity concentration, strength of waste drums, etc. Concerning waste drums to be received by Rokkasho LLW Disposal Center, JNFL operators inspects at nuclear power plants in advance whether those waste drums meet technical standards or not. After the inspection, drums are transported to Rokkasho LLW Inspection Building via a port near Rokkasho LLW Disposal Center. In LLW Inspection Building, JNFL operator checks outside surface and identification number of waste drums. After certificates of waste drums are given by a regulatory body, waste drums are transported to the No.1 or the No.2 Disposal Areas. These activities conducted by JNFL operators are indicated in guides or practices in the General Standards.

3.3. Quality Assurance for disposal facilities

As well as waste drums, the technical standards for disposal facilities are provided. The technical standards for disposal facilities consist of standards for disposal areas and standards for affiliated facilities. The technical standards for disposal areas consist of strength of bedrock, size and wall thickness of concrete pits, total radioactivity, specifications for cover soil, etc. Each technical standard is to be confirmed at proper stages by regulatory bodies. In order to satisfy these technical standards, all works concerning design, construction and operation conducted by JNFL have been standardized, and guides or practices mentioned in the General Standards have been established.

4. Outline of safety assessment of Rokkasho LLW Disposal Center

The No.1 and the No.2 Disposal Areas are to contain radioactive waste semi-permanently, so it is essential to assess the long-term radiological impact to the public. In this section, we shall describe the outlines of institutional control and safety assessment conducted to obtain the business license.

4.1. Institutional control in Rokkasho LLW Disposal Center

In Rokkasho LLW Disposal Center, contents of institutional control are to be relieved gradually, according to the decay of radioactivity included in disposed waste. This concept is called "Phased Control", and it is composed of three stages. After more than 300 years of institutional control period, the site is supposed to be used for general activities, such as agriculture, excavation, residence, etc.

4.2. Safety Assessment during institutional control period

In order to assess radiological impacts to the public, some representative exposure scenarios were selected at first, and then the dose values concerning these exposure scenarios were calculated.

Table I shows the result of the safety assessment concerning the No.1 Disposal Area. The scenario that would cause the maximum exposure dose value is the scenario caused by skyshine from waste, and the value is approximately 27 µSv/y. This value does not exceed the dose limit provided in the Japanese regulations (1 mSv/y), and radiological impact to the public is negligible.
### Table I. Results of Safety Assessment

<table>
<thead>
<tr>
<th>Period</th>
<th>Events</th>
<th>Scenarios</th>
<th>Dose (µSv/y)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Institutional Control</td>
<td>Normal Events</td>
<td>Inhalation of radioactive materials released in atmosphere from ventilation facilities</td>
<td>1.5E-03</td>
</tr>
<tr>
<td>Control Period</td>
<td></td>
<td>Ingestion of aquatic foods in Obuchi marsh that liquid waste flow in</td>
<td>4.4E-04</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Ingestion of aquatic foods in Obuchi marsh that groundwater flow in</td>
<td>3.1E-02</td>
</tr>
<tr>
<td></td>
<td></td>
<td>External and internal exposures at the valley that ground water flow in</td>
<td>4.1E-06</td>
</tr>
<tr>
<td></td>
<td></td>
<td>External exposure by skyshine from radioactive materials temporary stored or disposed</td>
<td>2.7E+01</td>
</tr>
<tr>
<td></td>
<td>Likely Events</td>
<td>Ingestion of aquatic foods</td>
<td>7.5E-02</td>
</tr>
<tr>
<td>After Institutional</td>
<td></td>
<td>Ingestion of surface water</td>
<td>1.3E-01</td>
</tr>
<tr>
<td>Control Period</td>
<td></td>
<td>Ingestion of agricultural products</td>
<td>9.1E-02</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Ingestion of livestock products</td>
<td>2.9E-02</td>
</tr>
<tr>
<td></td>
<td></td>
<td>External and internal exposures accompanied with agricultural work using irrigation water</td>
<td>5.5E-02</td>
</tr>
<tr>
<td></td>
<td></td>
<td>External and internal exposures accompanied with construction work of residence</td>
<td>8.3E-02</td>
</tr>
<tr>
<td></td>
<td></td>
<td>External and internal exposures accompanied with residence in a housing complex</td>
<td>1.5E+00</td>
</tr>
<tr>
<td></td>
<td>Unlikely Events</td>
<td>External and internal exposures accompanied with a large-scale excavation</td>
<td>8.1E+00</td>
</tr>
<tr>
<td></td>
<td></td>
<td>External and internal exposures accompanied with residence in a large building</td>
<td>1.4E+01</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Ingestion of well water</td>
<td>3.0E+00</td>
</tr>
</tbody>
</table>

### 4.3. Safety assessment after institutional control period

After institutional control period, restraint on some activities like excavation and residence will be lifted. Therefore, in the safety assessment after institutional control period, it is needed to consider assuming some scenarios such as ingestion of water, crops or dairy products, construction or residence of buildings. In addition, it is needed to consider the probability of occurrence of assumed scenarios, which is described in Japanese safety requirements for LLW disposal facilities.

The results of safety assessment shown in Table I represent the maximum exposure dose value is approximately 1.5 µSv/y for residence in a housing complex among likely events. Among unlikely events with lower probability of occurrence, the maximum is approximately 14 µSv/y for residence in a large building. These results of safety assessment do not exceed the dose limits provided in the safety requirements, and radiological impact to the public is negligible.

### 5. Summary

The Rokkasho LLW Disposal Center has been disposing of LLW from operating nuclear power plants in Japan. The total disposal amount of LLW has reached approximately 174,000 drums. To accomplish safe and reliable operation, JNFL has recently enforced quality assurance activities such as planning, design, operation, and maintenance. JNFL is intending to review and lighten the contents of institutional control gradually, according to the decay of radioactivity of disposed waste. To obtain the license of business and evaluate the safety of Rokkasho LLW Disposal Center, JNFL assessed the radiological impact to the public during the institutional control period and after the institutional control period. The results of the assessment revealed that dose values in the future would be negligible.
The current status of JNFL sub-surface disposal plan for relatively higher LLW

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Abstract. Since 2001, the investigation of geology and ground water in JNFL Rokkasho-mura site has been performing in order to acquire the basic data for the design and the safety assessment for the low-level radioactive waste (LLW) sub-surface disposal facility with engineered barrier. This paper briefly provides an overview and the current status of the project.

1. Introduction

The JNFL site is located in the legislated development area in Rokkasho-mura, Aomori Prefecture in the northern area of the Mainland of Japan, which is about 700 km away from Tokyo. The population density of Rokkasho-mura is 47 persons per square km and no population in the legislated area.

The surface of the site is about 7.5 km², where the Low-Level Radioactive Disposal Center, Uranium Enrichment Plant and High-Level Vitrified Waste Storage Center have been in operation and the Reprocessing Plant is in the midway of Uranium test.

The LLW is categorized in Japan according to the disposal concept as follows:

1) LLW for near-surface disposal without engineered barrier, e.g. extremely low contaminated waste as outside structure materials.
2) LLW for near-surface disposal with engineered barrier, e.g. the waste generated from operation and maintenance of power reactors.
3) LLW for sub-surface disposal with engineered barrier, e.g. relatively higher level waste than category 2).
4) LLW for geological disposal, e.g. highly contaminated waste and Trans Uranium (TRU) waste.

In Japan, the facilities of 1) and 2) above have already been in operation experimentally or commercially. JNFL has a plan of a new facility for category 3) LLW for sub-surface disposal with engineered barrier, for short “the sub-surface disposal facility”. This facility can accept the relatively higher low-level waste from power plant operation and in core materials from the decommissioning, estimated to be about 20 thousands cubic meter in total. In addition, this will accept TRU slightly contaminated waste from reprocessing plant operation and decommissioning.

In 2001, the preliminary survey had been conducted to estimate the possibility to construct the facility at Rokkasho site of JNFL. Since 2002, investigation of geology and groundwater in detail has been being performed in order to collect various data for the conceptual design.

2. Operation of ROKKASHO LLW Disposal Center

At Rokkasho LLW Disposal Center, such low-level waste as generated from operation of domestic nuclear power plants have been accepted and buried since 1992. There are two facilities called No.1 and No.2, whose capacity is 40,000 m³, respectively. The disposal facilities are on a terrace some 30 to 60 meters above sea level, and each one is separated in dozens of concrete pits, installed in excavated mass of low permeability bedrock about 15-20m below the surface. The No.1 disposal facility had accepted homogeneous waste including cemented or bituminized condensed liquid waste, spent resin and so on. The waste buried in No. 2 disposal facility, operating since 2000, is solidified activated metals and plastics waste, etc with mortar. By the end of this February, 135,899 waste packages have been disposed of in No.1 facility and 36,352 in No.2 in the shape of 200-liter drum.
3. Concept of the sub-surface disposal in Japan

3.1. The targeted waste

The facility under planning will accept such wastes whose concentration levels can be a few order higher than that disposed of in current facilities in Rokkasho site. Proposed waste contains the following ones:

- the reactor core surrounding parts i.e. the channel boxes, control rods and burnable poisons,
- some of the ion exchange resin with highly condensed reactor sludge,
- the reactor core-internals replaced or decommissioned,
- some of the waste from reprocessing plant operation and decommissioning, which will contain low-level TRU elements.

As acceptable waste packages, 200-liter waste drums and 1-5 m$^3$ square metallic containers are considered.

3.2. The basic concept of sub-surface disposal in Japan

In 1998 the Atomic Energy Commission of Japan reported that the sub-surface disposal facility requires (1) enough depth not to restrict general use of underground i.e. 50-100m depth, (2) no existence of natural resources, (3) sufficiently long groundwater pathway and (4) preservation and openness of records on the repository. The facility should be under control by implementer during several hundred years, through monitoring of leakage of nuclides and restriction of land-use. After several hundred years, some measures should be taken to prevent human intrusion, the unsafe exposure, and to preserve adequate groundwater pathway. And it also stated, “When the license is over, the licensee hands the records of the facility to the government. And the records of place, waste kind and volume and nuclide concentrations can be preserved forever”.

4. The plan and the current status of the JNFL sub-surface disposal facility

4.1. General plan

The preliminary survey had been performed from 2001 to 2002. The detailed investigation has succeeded them for about 3 years. Based on the results the design of the facility will be defined and the documents on safety assessment will be served for the safety review by the coming safety standard. At the same time, it is essential to get public acceptance in the local communities. In general, it will take a few years to pass the safety review and several years to construct the facility.
4.2. The preliminary survey

From 2001 to 2002, the preliminary survey had been conducted to acquire data to assess the possibility for construction of the facility (See fig.1). The survey contains geology survey, e.g. 9 borings and elastic wave testing, and groundwater survey, e.g. permeability tests and geo-chemical tests.

The Takahoko layer, the new tertiary deposit in Miocene epoch, spreads widely at the depth of 50-100m below ground level in the site. It consists of both of tuff and sandstone in the middle layer, and mudstone in lower layer. It is confirmed that it has few cracks and is strong enough to excavate tunnels by triaxial compression tests. Moreover, it is found that the groundwater flow is enough slow by permeability tests and movement of the nuclides will be limited. Most of water from the precipitation runs off the surface or passes through the quaternary deposit of the surface to the surrounding of the terrace. A portion of it penetrates into Takahoko layer and flows out slowly in the deep area and the surroundings of the terrace. As the result of the preliminary survey, it is determined that the facility could be installed there.

4.3. The detailed investigation

From 2002 to 2005, the detailed investigation of geology and ground water has been performed in order to collect various data for the safety review. To confirm hydraulic conditions and geo-chemistry, 13 boring surveys including 6 holes in swamp and marsh have been performed.

The excavation of the 1 km long access tunnel (the entrance level EL 8.0m, incline of 1/10) to the altitude of EL -86m underground has just been finished. During excavating the tunnel, observation of geology, physics tests, permeability tests, pore water pressure measurements and so on had been performed in situ. And the large size test cavern has been under construction at the end of the tunnel to demonstrate stability of the disposal facility (see Figures 2 and 3). Prior to the excavation, three measuring tunnels were excavated surrounding the test cavern to examine the excavation.

4.4. The concept of the facility

Two types of the sub-surface disposal facility exist in the world, i.e. a tunnel type such as SFR repository in Sweden and a silo type such as VLJ repository in Finland. Considering the ground condition of EL -90m underground at the site, a tunnel type one will be more realistic. The diameter of the tunnel will be led out by the results of detailed investigation. In the tunnel, the concrete pit will be installed, which works both as a basement and as crane abutment for placing the laden waste packages. The multiple engineered barrier system can consist of the metallic containers, the filler (a low diffusion zone), e.g. cement, the bentonite (low permeability zone) and the backfill (see Figure 4). The host rock works not only to support the engineered barriers but also as the natural barrier and the hydraulic condition and geological properties of the natural barrier have been under investigation in detail and analyzed.

![FIG. 2: Entrance of the access tunnel](image)
5. Prospect

The conceptual design of the facility is being carried out as well as on going detailed investigation. For assuring safety of the sub-surface disposal, it is necessary to predict precisely the function of engineered barriers and natural barrier. To prepare the safety review documents it will be essential to construct safety case of the repository based on the results of the detailed investigation and referring to the results of studies in the geological disposal. At the same time, the guideline for the safety review is necessary to be prescribed by the Nuclear Safety Commission of Japan, and the safety regulation scheme is to be built by the government.

To develop this project is to take a big pace forward to attain overall disposal system of LLW along with the current waste disposal technologies. This project will contribute to implement the policy of the waste management and to attain safety disposal of miscellaneous wastes generated from fuel cycle facilities and other sources in Japan.
Condition monitoring and service life prediction of near surface disposal module located at three different sites in India

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India

Abstract. In India, reinforced concrete trenches (RCT) are used extensively for disposal of low and intermediate level solid radioactive waste in near surface disposal facilities (NSDF), co-located with nuclear power plants at seven different sites in the country. In order to study the performance of RCTs, an extensive programme was undertaken for three representative sites located at Rajasthan (inland) and Trombay & Tarapur (coastal), which are under operation for last three to four decades. Central Building Research Institute (CBRI) a national research organization was engaged in this study in view of their in-depth expertise in design of civil structures, condition monitoring and remediation. The condition monitoring of RCTs was done using various non-destructive techniques to estimate strength of concrete and probability of corrosion of reinforcement bars. Concrete was also characterized for chemical properties to test the depth of carbonation and chloride penetration in the RCT. Besides this, effect of soil and groundwater on RCT was taken into consideration. Mathematical model and computer code based on reliability theory was developed and validated to predict the service life of trenches. Probability of failure for different target life has been calculated for different values of safety indices. In this paper, details of condition monitoring and residual service life prediction are presented.

1. Introduction
The safe disposal of radioactive waste aims to isolate the waste so that it does not result in undue exposure to human and environment. Near surface disposal represents an option commonly used to contain low and intermediate level waste (LILW). The disposal facility is located on or below ground level with thick protective covering. Near surface disposal is being practiced in India for last three to four decades at various sites. Though there is a wide variation in site characteristics, types and amounts of waste and in the design of facilities, the experience has shown that effective isolation of waste depends on the performance of overall disposal system comprising the waste form as well as engineered and natural barriers. Among these, the performance of engineered barrier through its designed life plays a key role in disposal and isolation of LILW. Reinforced concrete trenches are used extensively in these NSDF in India. In order to study the performance of RCTs an extensive programme of condition monitoring and service life prediction was undertaken for three representative sites in the country. The study of reinforcement corrosion in concrete structures, its monitoring and service life prediction has also been carried out [1-4].

2. Reinforced concrete trenches
At the disposal site, RCTs are laid out zone wise and are constructed in batteries. The dimensions of the trench vary from site to site. In general, these trenches are 1.8 to 4.8m deep, 1.2 to 3.1m wide and 6 to 15m long. The external walls are 180 to 450mm thick, the internal walls are 165 to 350mm thick and the raft is 250 to 600 mm thick. The trenches are constructed in batteries with width of batteries ranging from 8 to 20m and length ranging from 15 to 50m. The RCTs are normally built underground, with only roof above ground. Typical cross section of RCT battery is shown in Fig.1.

These RCTs are designed as per the principles of underground water tank following the codes IS: 3370 [5] & IS: 456 [6]. The grade of concrete used were M15 (f_{ck}=15 MPa) and above. As a part of waterproofing two layers of stone tiles laid below the raft in cement sand mortar with waterproofing (w/p) compound. On outer face of external walls, one layer of stone tiles is fixed in cement sand and w/p compound mortar at corners of tile. The joints between two tiles are pointed with the same type of mortar. The gap between the wall and tiles filled with cement sand and w/p compound slurry and
15mm thick finishing coat over tiles with same mortar is done. Detail-A of Fig. 1 shows the section of wall with waterproofing layer. The size of tiles and width of joints vary with respect to site. As a part of closure programme, pre-cast concrete covers are fixed in top notches of trench and top waterproofing done with suitable slope.

![Diagram of Typical cross section of RCT](image)

**FIG. 1: Typical cross section of RCT**

3. **Condition monitoring of RCT**

3.1. **NDT of concrete with rebound hammer**

The waterproofing layer over concrete was removed before performing different NDTs. The rebound hammer test described in IS: 13311 (Part-2) [7], ASTM C805 [8] were performed using Schmidt Rebound Hammer. The tests were performed on old as well as on new trench at each site. The results in brief are shown in Table 1 [9].

<table>
<thead>
<tr>
<th>Site</th>
<th>RC Trench</th>
<th>Range of compressive strength (N/mm²)</th>
<th>Average Compressive strength (N/mm²)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rajasthan</td>
<td>Old (No. 1)</td>
<td>18.26 to 36.32</td>
<td>26.19</td>
</tr>
<tr>
<td></td>
<td>New (No. 11)</td>
<td>9.0 to 42.21</td>
<td>30.10</td>
</tr>
<tr>
<td>Trombay</td>
<td>Old (No. 19A)</td>
<td>18.68 to 25.2</td>
<td>20.54</td>
</tr>
<tr>
<td></td>
<td>New (No. 38P)</td>
<td>18.68 to 27.86</td>
<td>22.85</td>
</tr>
<tr>
<td>Tarapur</td>
<td>Old (No. E1/E2)</td>
<td>21.35 to 29.46</td>
<td>26.47</td>
</tr>
<tr>
<td></td>
<td>New (No. GG5/HH5)</td>
<td>20.9 to 36.49</td>
<td>25.58</td>
</tr>
</tbody>
</table>

3.2. **NDT of concrete with ultrasonic pulse velocity**

A Portable Ultrasonic Non-destructive Digital Indicating Tester (PUNDIT) was used for carrying out ultrasonic pulse velocity (USPV) test as described in codes [10, 11]. In this technique, since there is no unique correlation between the velocity and strength of the concrete of different mix proportions, this technique is used more as a qualitative test.

Five point scale given by A.W.Neville [12] for quality assessment of concrete vis-à-vis pulse velocity is shown in Table 2. The results are presented in Table 3.
Table 2: Quality assessment of concrete vis-à-vis pulse velocity [12]

<table>
<thead>
<tr>
<th>Pulse velocity (Km/sec)</th>
<th>&gt;4.6</th>
<th>3.5 – 4.6</th>
<th>2.9 – 3.5</th>
<th>2.0 – 2.9</th>
<th>&lt;2.0</th>
</tr>
</thead>
<tbody>
<tr>
<td>Quality of concrete</td>
<td>Excellent</td>
<td>Good</td>
<td>Fair</td>
<td>Poor</td>
<td>Very poor</td>
</tr>
</tbody>
</table>

Table 3: Results of ultrasonic pulse velocity tests

<table>
<thead>
<tr>
<th>Site</th>
<th>RC Trench</th>
<th>Quality of concrete</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rajasthan</td>
<td>Old (No. 1)</td>
<td>Good to excellent</td>
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<tr>
<td></td>
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<td>Good</td>
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<td>Trombay</td>
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<td>Tarapur</td>
<td>Old (No. E1/E2)</td>
<td>Excellent</td>
</tr>
<tr>
<td></td>
<td>New (No. GG5/HH5)</td>
<td>Fair</td>
</tr>
</tbody>
</table>

3.3. NDT for detecting corrosion of reinforcement by Half-cell potential method

Multi-Cell surveyor was used to assess the corrosion of reinforced bars in the trench and this method is called half-cell potential method. The corrosion of steel in concrete is an electrochemical process, which develops anodic (corroding) and cathodic (passive) sites on the steel surface. The effective measurement of potential between these would give possibility of corrosion. Interpretation of half-cell potential test results is carried out as per the ASTM C876 [13] guidelines, and is presented in Table 4.

Table 4: Interpretation of half-cell potential results as per the ASTM C876 [13]

<table>
<thead>
<tr>
<th>Half-cell potential (mV)</th>
<th>&lt;–350</th>
<th>–200 to –350</th>
<th>&gt;–200</th>
</tr>
</thead>
<tbody>
<tr>
<td>Percentage chance of active corrosion</td>
<td>90</td>
<td>50</td>
<td>10</td>
</tr>
</tbody>
</table>

3.4. Laboratory tests

Soil samples and groundwater were collected from the three NSDF sites and were analyzed in the laboratory. Soils were analyzed for grain size distribution, plasticity indices, soluble salts, pH, etc. where as groundwater was analyzed for pH, soluble constituents and major ionic species present in it. Samples of concrete (cores) collected from RCT of three sites were also analyzed for pH, soluble salts and alkalinity and their compressive strength was determined as per BS: 1881 (Part 120) 1981 [14]. The results of the laboratory tests are discussed under section 5.0.

4. Service life prediction of RCTs

The service life of the reinforced concrete structure is composed of two parts, viz. (i) the initiation period i.e. time period from construction to the start of corrosion in reinforced steel and (ii) propagation period i.e. time period between reinforcing steel start to corrode to spalling of concrete.

Several researchers [1, 2, 3] have presented different mathematical models for service life prediction. The major challenges in developing a mathematical model for service life prediction is that, many parameters including material characteristics, climatic environment and construction method will all affect the service life and it becomes difficult to specify many of these parameters with sufficient precision for them to be used in a mathematical model.
4.1. Mathematical modelling

The durability models for chloride penetration and carbonation process are used for calculating mean service life of repositories due to chlorination and carbonation. Details of these models are as below;

4.1.1. Chloride induced corrosion

As a result of chloride penetration, a gradient develops near the concrete surface. The time, at which the critical chloride content reaches the steel surface and depassivate it, can be considered as initiation time of corrosion. The gradient of chloride content is often described by an error function model, which fulfils the condition of Fick’s second law of diffusion. This model is very well proved in field and laboratory tests. The formula for initiation time of corrosion can be written as,

$$t_p = \frac{1}{12 \times D} \left(\frac{C}{1 - \left(C_{cl} / C_l\right)^{1/2}}\right)^2$$

where,

- $C_{cl}$ = Critical chloride content
- $C_l$ = Chloride content at concrete surface
- $D$ = Diffusion coefficient
- $C$ = Concrete cover

4.1.2. Carbonation induced corrosion

A service life relation based on carbonation of concrete, as given by Vrouwenvelder[15] is shown below. The first term of this equation describes the actual carbonation of concrete cover without waterproofing layer. The second term describes the time after initiation of crack. The last term is increased service life due to use of protective coating i.e. waterproofing.

$$L = \left[\left(\frac{C - \Delta}{R \times K} \left(\frac{0.027}{46w - 17.6}\right)\right)^2 + \frac{C \times c}{\phi \times V_c} \left(\frac{T}{T_0}\ln f_0\right)\right] \times S$$

where,

- $\Delta$ = difference of max and mean depth of carbonation (mm)
- $R$ = Cement type parameter
- $w$ = water cement ratio
- $\phi$ = bar dia (mm)
- $S$ = thickness of coating (mm)
- $f_0$ = coefficient of imperfection for coating
- $T_0$ = durability parameter (years)
- $T$ = frequency of maintenance (years)

4.1.3. Service life prediction using log-normal distribution

All the parameters involved in the durability analysis viz. environmental action, deterioration process and material behaviour are random in nature. The problem of service life estimation has been solved by assuming distribution of service life as log-normal distribution. The theory of log-normal service life distribution was first elaborated for durability problems by Siemes, Vrouwenvelder and V.D.Beukel [16]. Based on this theory a computer program was developed to predict mean service life due to carbonation and chloride ion attack. The program computes mean service life, standard deviation of service life and probability of failure for different target lives of repositories.
5. Conclusions

The salient outcomes of present studies are:

(a) Results of NDTs show that strength of concrete of various RCTs at three NSDFs is well above the design strength and the quality of concrete in trench is still very good after 30-40 years.

(b) Laboratory study reveals that the soils are not having appreciable swelling characteristics to cause structural distress in trenches. With mildly alkaline nature and low proportion of water-soluble salts, the soil and ground water not expected to cause any appreciable chemical deterioration in the structure. Concrete samples from sites are still having moderate level of alkalinity, indicating resistance to corrosion.

(c) Probability of corrosion of reinforcement was found to be negligible in new RCTs. In old trenches at Trombay and Tarapur, increase in probability of corrosion with depth was noticed. Physical examination of re-bar has shown that the extent of corrosion is negligible.

(d) Probability of failure for different target life has been calculated for different values of safety index. The programme has been validated with published research data.

(e) From parametric study, it was found that water-cement ratio, cover thickness and width of joint between tiles are the major parameter having significant influence on the service life.

(f) In all three sites, surface chloride content is far below the critical chloride content (0.16% of cement content). Hence, the failure of structure due to chloride ion attack is not expected.

(g) Considering the various parameters of six repositories of three sites, which have been investigated, RCTs will have a service life of about 240 years with proper maintenance at regular intervals.

REFERENCES

Safety of radioactive waste disposal during Enmasee coolant channel replacement of Indian PHWR

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Abstract: Pressurized Heavy Water Reactors (PHWR) used in India have demonstrated high reliability during the several decades of reactor years of operation. One of the life limiting components of the reactor is the coolant channel assembly, which is designed to be replaced. The pressure tubes operate in the environment of high neutron flux, high temperature and high pressure. This environment affects the service life of the pressure tubes (PT). The limits which influence the in-service life of PT in the reactor are dimensional changes due to creep, induced neutron/hydride embrittlement, fatigue, thermal gradient caused at the point of contact of PT with Calandria Tube due to garter spring movement, which could result in blisters and lead to failure of PT. Thus it is necessary to replace the coolant channel assembly of the reactor after operating it for designed reactor years of operation. In India to date three reactors had undergone the Enmasse Coolant Channel Replacement Campaign (ECCR). The handling and disposal of radioactive waste generated in such campaigns need to be planned properly taking into consideration radiological and other safety aspects. This paper mainly focuses on the types of radioactive wastes generated during such campaigns, further treatment of such wastes and the various disposal options adopted for enhancement of safety. It also covers the strategic changes/improvements in the design of size reduction system, disposal facilities in three ECCR campaigns carried out in India mainly for safe handling of radioactive waste with minimum man-rem expenditure. Further strategies, which could be adopted in future for volume reduction of disposable waste, are also briefed.

1. Introduction
Pressurized Heavy Water Reactors (PHWR) having 235 MWe of installed capacity each are located at various sites in India. All these reactors use natural uranium as fuel and heavy water as the moderator and primary coolant. These reactors have 306 horizontal coolant channel assemblies each of which has 12 fuel bundles with shielding plugs and sealing plugs at either ends. The pressure tubes of these reactors are made of Zircalloy-2. PHWRs in India have demonstrated high reliability during the several decades of reactor years of operation. One of the life limiting components of the reactor is the coolant channel assembly, which is designed to be replaced.

2. Need for Enmasee coolant channel replacement
The pressure tubes operate in the environment of high neutron flux, high temperature and high pressure. This environment affects the service life of the pressure tube (PT). These pressure tubes are supported on end shields by the end fitting and the two garter springs ensure the gap between PT and calandria tube. During the extended operation, the deuterium level increases in pressure tube which are made of Zircalloy-2. This increase in deuterium level also makes the PT more susceptible to delayed hydrogen cracking. The limits which influence the in-service life of PT in the reactor are dimensional changes due to creep, hydrogen pick up, thermal gradient caused at the point of contact of PT with calandria tube due to garter spring movement, which could result in blisters and lead to failure of PT. Thus, it is necessary to replace the coolant channel assembly of the reactor after operating it for designed reactor years of operation. In India to date three reactors had undergone the Enmasse Coolant Channel Replacement Campaign (ECCR) namely Rajasthan Atomic Power station –Unit 2 (RAPS-2), Madras Atomic Power Station-Unit 2 (MAPS-2), Madras Atomic Power Station-Unit 1 (MAPS-1). The handling and disposal of radioactive waste generated in such campaigns requires meticulous planning and concentrated efforts due to high radiation fields, large quantities and odd dimensions of the components requiring creation of additional facilities for their handling, transport, cutting, sizing, conditioning and disposal. The various types of solid metallic waste generated and disposed are detailed in Table 1.
Table 1: Types of radioactive wastes generated during the Enmasse Coolant Channel Replacement Campaign (ECCR).

<table>
<thead>
<tr>
<th>Sl no</th>
<th>Item</th>
<th>Qty</th>
<th>Dimensions</th>
<th>App Weight (Te)</th>
<th>Curie Content after cooling period</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>No.s</td>
<td>Volume (m³)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1.</td>
<td>Pressure tubes (PT) with end shield plugs</td>
<td>306</td>
<td>8.97</td>
<td>5.2mx9cm</td>
<td>11.47</td>
</tr>
<tr>
<td>2.</td>
<td>End fittings (EF)</td>
<td>612</td>
<td>60</td>
<td>2.1mx15cm</td>
<td>84.2</td>
</tr>
<tr>
<td>3.</td>
<td>Garter Springs</td>
<td>612</td>
<td>1</td>
<td>015cmx9cm</td>
<td>0.10</td>
</tr>
<tr>
<td>4.</td>
<td>Assorted waste</td>
<td>1000 drums</td>
<td>200</td>
<td>-</td>
<td>200</td>
</tr>
<tr>
<td>5.</td>
<td>Combustible waste</td>
<td>50 drums</td>
<td>10</td>
<td>-</td>
<td>1.4</td>
</tr>
</tbody>
</table>

RAPS-2 was the first Indian unit to undergo the ECCR programme. This demonstrated the ability and created the confidence to carry out such major core component replacement works in the country. The tooling and processes that were used at RAPS were simple mechanical tools with minimum automation, which took considerable time for carrying out the campaign. The size reduction of the coolant tube (i.e. cutting the tubes in to equal parts) during this campaign was carried out with the help of a hacksaw-cutting machine enclosed in a shielding chamber.

3. Radioactive waste management during Enmasse coolant channel replacement of MAPS-II

MAPS-2 was the next reactor to undergo the ECCR campaign in year July 2002 after completing designed full power years of reactor operations. The experience gained at RAPS-2 has been effectively utilized in modifying the tooling and procedures for safe handling of the radioactive waste. A small step was taken towards automation during this campaign. Since the cooling period for the irradiated reactor components after defuelling and dilute chemical decontamination was only 120 days, the disposal of these waste materials posed a real challenge due to high radiation fields, large quantities and odd dimensions of the components. The management strategy planned for the disposal of these components included cutting of these components to proper size at the disposal site, where a suitably designed and shielded system was utilized.

3.1. Disposal of end fittings

During this campaign the end fittings shipping cask itself was used for carrying out the disposal operation. The endfittings were taken out from top of the cask and lowered into the tile hole remotely using crane. This operation was carried out from a distance and by erecting a tent around the disposal spot to avoid spread of contamination. This was followed by embedding the end fittings in concrete and covering tile holes with precast concrete caps to bring down the radiation fields above the tile holes below the prescribed limits.

3.2. Conditioning and disposal of pressure tubes

Since, the pressure tubes were of full length (5.2 meters long), it required cutting at the disposal site to match the disposal facility. The cutting of the pressure tube in to two equal parts was done by a chipless tube cutting machine having following salient / safety features:

- Machine achieves cutting by a rotary orbital cutter which causes fine indentions of the tube and cutting is performed by progressive indentation and tearing of the tube. The cutting system was housed in the shielded enclosure so as to reduce the radiation level below the prescribed limit.
- Complete shipping cask / disposal cask docking with machine enclosure, tube feeding, cutting system were automated and encased in a casing to reduce spread of contamination.
- Minimum generation of secondary waste / air-borne activity during cutting of tube.
- Remote viewing through cameras and lead glass windows.

The pressure tubes after cutting were remotely collected / transferred to disposal cask through which the tubes were disposed into the tile holes using doughnuts. This was followed by embedding the
pressure tubes in concrete and covering tile holes with precast concrete caps to bring down the radiation fields above tile holes below prescribed limits.

4. Radioactive waste management during Enmasse coolant channel replacement of MAPS-I

After gaining the experience from ECCR programme of MAPS-II, several steps were taken to bring the manrem intensive activities to a semi-auto mode of operation, increased mechanization in the areas of radioactive material transfer and handling system.

4.1. Disposal of end fittings

In the previous campaigns the end fittings were taken out of the cask into open air and disposed in the tile hole by erecting a tent around the disposal spot. In MAPS-1 campaign, the end fittings were directly disposed from end fitting shipping cask into the tile holes with the help of specially developed end fitting disposal system, without bringing the end fitting in open air. Thus considerably saving in the man-rem expenditure during handling and disposal of end fittings.

4.2. Under water storage of pressure tubes

As an alternative for cutting and disposal of pressure tubes and to facilitate the further volume reduction at later date, underwater storage of the pressure tubes with requisite surveillance was planned. The change in strategy for storing the pressure tubes called for challenges in designing of mechanical systems such as split type pressure tube transfer cask, under water tube handling and storage rack system. The handling of pressure tube at storage site mainly involved operations like transferring of pressure tube from shipping cask to split type tube transfer cask, underwater handling of tube transfer cask, underwater transfer of the pressure tube from the split cask to the underwater storage rack with the help of pneumatically operated grippers mounted on tube handling carriage. All the equipments like handling units and casks were designed, tested and detailed safety analysis were carried out before taking up the campaign. (Fig. 1 Underwater Storage of Pressure tubes).

The focus of the MAPS # 1 campaign was:

a. low down time;
b. reduction of manrem;
c. adopting new Technology;
d. better material handling facility for enhanced safety of operating personal;
e. training and certification for all Personnel;
f. effective planning and control.

These changes have resulted in the campaign being safely completed in a record time and with reduced man-rem expenditure setting a new benchmark.

5. Volume reduction of pressure tubes

As a part of management strategy, pressure tubes are stored in water pool for a limited period of time to bring down the activity/radiation level and will undergo size/volume reduction before final disposal. Salient features of this scheme are as follows:

- Tubes stored in pool will be retrieved using existing infrastructure available such as gantry crane, split cask, tube handling carriage, yokes etc.
- Volume reduction of the tubes will undergo two stage treatment.

a) **Size reduction** i.e. cutting of tube into small pieces of size 200mm using hydraulically operated shearing machine. These pieces will be filled into standard 200 litre drums.

b) **Compaction of drum by hydraulically operated compactor.**

The standard 200 litre drum containing coolant tube pieces/coolant tubes will be compacted to form a pellet which will be collected in M.S container for safe disposal. All the above operations are planned and designed to be carried out remotely with minimum manual intervention considering all the safety aspects.
6. Disposal of radioactive waste generated in coolant channel replacement campaign

A multibarrier approach is followed in disposal of radioactive solid waste depending on activity of waste. The overall safety against migration of radionuclide is achieved by proper selection of waste form; suitable engineered barrier and the characteristic of the geo-environment of the site. Waste of low activity like assorted wastes are disposed into the stoned lined trenches. Combustible wastes are incinerated and ash generated is collected into the drums, immobilized into cement matrix and disposed as solid waste in trenches.

The highly active solid waste generated in the coolant channel replacement campaign i.e end fittings, garter springs and pressure tubes etc are disposed into the Reinforced Concrete Cement (RCC) tile hole. Tile hole are underground circular vaults of 710mm diameter and a depth of 4 m.

It is made out of carbon steel shell with one end closed and is lined by spinning cement mortar with necessary reinforcement cage. As an additional protection waterproofing tiles are provided on the outer surface. Each tile hole is provided with a 500mm thick RCC plug on the top for shielding purpose during operation and after permanent sealing of tile hole. A detail of tile hole is shown in Fig.2. RCC tile holes are the engineered structures constructed to minimize or prevent the water ingress so as to lessen the leaching of the radioactivity to environment and act as secondary barrier.

After the disposal facility is filled, it is required to be sealed / closed to minimize the contact with rainwater, avoid exposure of waste to environment and its subsequent dispersal to environment, minimize undue exposure. In case of tile hole the practice followed is to fill the voids between the wastes with cement grout after disposal of the solid waste. This makes a tile hole monolith of concrete. The top of tile hole is covered with concrete plug to avoid rainwater entry and bring down radiation field to minimum prescribed limit.

Provision for monitoring and periodic surveillances is incorporated during the design of the disposal facility itself. Boreholes of 6 to 20m deep are provided at appropriate locations. The ground water levels and water samples are monitored periodically. Soil and vegetation samples from the site are also periodically collected for any uptake of radioactivity. Radiation survey of the entire site is carried out at predetermined intervals. The entire disposal site is totally closed by a physical security wall to avoid unauthorized access.
Radioactive contamination of soils in Belarus: experience and trends of rehabilitation

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Abstract. The paper presents the analysis of the developed and suggested measures and methods on decreasing radionuclide soil contamination in the Republic of Belarus after the Chernobyl accident and prevention of migration of the long-lived radionuclides $^{137}$Cs and $^{90}$Sr in food chains. It was found that the application of protective measures based on decreasing of $^{137}$Cs and $^{90}$Sr mobility in soil due to their fixation in soil-absorbing complex, or so-called rehabilitation, is the more effective, ecologically safe and, it seems, the most prospective countermeasure in Belarus at present. The possibility of applying fresh water lake sediments, tripoli and industrial waste for rehabilitation of contaminated soils and prevention of $^{137}$Cs and $^{90}$Sr accumulation in agricultural production in Belarus are considered.

1. Introduction

After the Chernobyl NPP accident the methods and techniques on reducing of soil radioactive contamination and prevention of long-lived radionuclide migration, mainly $^{137}$Cs and $^{90}$Sr, in the system ‘soil–plant’ can be conditionally divided into two basic groups. The first group includes cardinal measures on total or partial extraction of radionuclides from the soil, i.e. decontamination. To the second group it is necessary to refer the protective measures on conditioning for reducing of radionuclide mobility and further migration in the system ‘soil–plant’.

Under the assessment of the Chernobyl NPP accident consequences on ecosystems of Belarus it should be taken into account that in the Republic more than 1,35 million ha lands are now contaminated by $^{137}$Cs with the density of 37—1480 kBq/m$^2$, from which 478,000 ha are contaminated by $^{90}$Sr with the density of 1—11 kBq/m$^2$ are used for agricultural production [1]. Thus, at minimizing the consequences of the Chernobyl accident in Belarus the main task remains development of complex measures on rehabilitation of agricultural soils contaminated by $^{137}$Cs and $^{90}$Sr with the purpose of preventing radionuclide migration in biological chains and decreasing of their intake into agricultural production.

Decontamination of soils contaminated by radionuclides can be conducted with two methods including mechanical, chemical and biological methods, and their combinations as well:

1. using mechanical removal of a upper soil layer contaminated by radionuclides or deep ploughing with a turnover of an upper soil layer;
2. using removal of radionuclides from soil by their chemical leaching or with application of biotechnology.

The protective measures are based on two principles:

1. conditioning for prevention of radionuclide absorption by plants – as protective measures;
2. conditioning for radionuclide fixation in soil – so-called rehabilitation.

In the initial period of conducting soil decontamination after the Chernobyl accident in Belarus, the mechanical removal of the upper contaminated layer of soil was widely conducted with the help of a total removal of soil or deep ploughing with a turnover of an upper layer. However by using this method it was necessary to restore humus layer of soil with the help of inserting a great deal of organic and mineral fertilizers in soil or to replace deleted contaminated soil with pure fertile soil entirely.
The chemical leaching of radionuclides from soil can be used at fixing surface contamination immediately after radionuclide fall-out by means of covering of soil surface with special emulsions with consequent removal of the hardened layer together with radioactive contamination.

The chemical rinsing of soils can also be applied for radionuclide removal except for contamination by a broad spectrum of radionuclides. The greatest effect in application of mechanical and chemical methods of soil decontamination will be reached at most high levels of contamination. It is necessary to mark that the application both mechanical and chemical methods of soil decontamination is perspective only in case of local contamination and it requires additional measures on restoring soil fertility.

As it is known soil decontamination is laborious, rather expensive (up to 1000 dollars for 1 m³ soil taking into account expenses of removal of the contaminated soil, its transportation and preparation) procedure, which requires a complex approach, but it is difficult to realize in the present economic conditions of the Republic of Belarus. The technical realization of such works in the most contaminated regions of Belarus is especially difficult. A high cost of technical-engineer measures and absence of effective methods of decontamination of natural landscapes have resulted in making decision on development of complex protective and rehabilitation measures directed to decreasing of radionuclide intake in vegetative bio-mass and agricultural production already at the first stage of liquidation of consequences of the Chernobyl accident in Belarus.

2. Decontamination and rehabilitation methods, and practice

At the present time in the agriculture of Belarus in the radioactive zone the protective and rehabilitation measures are carried out for obtaining of normatively pure production. These countermeasures directed mainly to obtaining of agricultural production with normative-permissible radionuclide content provide for optimization of physicochemical properties of soils by liming, inserting organic fertilizers and overdoses of mineral fertilizers as well [2].

As organic fertilizers cow manure is mainly applied. It is considered that depending on climatic and soil conditions these countermeasures can considerably reduce contamination of vegetable production. The positive effect is reached both due to increasing of migration of radionuclides ¹³⁷Cs and ⁹⁰Sr into plants and due to crop productivity increasing – so-call ‘dilution effect’. However the last scientific data show low efficiency of the countermeasures mentioned above and some of them are not safe from the ecological point of view. Permanent inserting of overdoses of mineral fertilizers causes anxiety in particular concerning ecological situation.

Using of a principle of decreasing of ¹³⁷Cs and ⁹⁰Sr mobility in the soil due to their fixation in the soil-absorbing complex, so-called rehabilitation, is effective enough, ecologically safe and, apparently, the most perspective measure of prevention of radionuclide migration in the system ‘soil-plant’.

As the practice displays the application of cow manure as amendments is an effectual measure of decreasing of radionuclide contamination of agricultural production in Belarus. However in the conditions of Belarus the application of cow manure and fertilizers on its basis is connected with a range of technical difficulties. At first there is no large quantity of cow manure. Secondly it is necessary to insert pure cow manure, i.e. it is necessary to bring it from the uncontaminated areas. But in Belarus there is no industrial production of organic fertilizers (first of all cow manure recycling), its transportation and insertion it in the untreated state is expensive and technically difficult problem.

Besides it is necessary to take into account that application of cow manure is effective mainly in the relation to ⁹⁰Sr, while for ¹³⁷Cs mobility decreasing it is necessary to apply other mineral substances and materials, which have high sorption properties in the relation to this radionuclide.

With this purpose along with using of such famous in radiochemistry radionuclide adsorbents as clay minerals and zeolite a range of authors suggested using bottom sediments of freshwater lakes (sapropels) widely for decontamination and rehabilitation of soil contaminated by radionuclides [3, 4, 5]. Thus, development of new types of effective and technically acceptable amendments on the basis of organic and mineral raw materials is rather promising for rehabilitation of soils contaminated by
radionuclides in Belarus. The sorbents should be ecologically safe and their properties should approximate to a natural fertile layer of rehabilitated soil.

A range of authors hold the opinion that inserting of sorbents selective to the radionuclides and mixed sorption materials and substances obtained from local organic and mineral raw materials of the Republic of Belarus is a very perspective agrochemical measure to solve problems on soil decontamination and rehabilitation in Belarus.

3. Perspective natural materials and substances

At the present stage of minimization of consequences of the Chernobyl accident the amendments on the basis of local natural organic and mineral raw materials – bottom sediments of freshwater lakes (sapropels), tripoli and industrial waste (hydrolyzed lignin and clay-saline slame) are of great interest for rehabilitation of soils contaminated by radionuclides in the Republic of Belarus [7, 8].

More perspective organic-and-mineral amendments in agriculture of Belarus are freshwater lakes sediments called “sapropel”. The basic advantages of sapropels as possible amendment to soil for radionuclide fixation are a high degree of dispersion, a great content of organic matter (up to 70 %) and the content of valuable nutrient substances. The major agrochemical properties of sapropels are determined with the availability of the mobile forms of basic nutritious elements. It is pointed out that the efficiency of sapropels is also determined with the quality of organic substances, especially of humic acids and nitrogen. The maximum quantity of nitrogen reaches 4,6 % per dry substance, and its basic quantity enters the composition of amino acids.

The Republic of Belarus has more than 1759,1 million m$^3$ sapropel resources [9]. The main types are the following: organic, carbonate, silica and the mixed types (Table I).

### Table I: Resources and typological breakdown of sapropel sediments in Belarus lakes

<table>
<thead>
<tr>
<th>Region</th>
<th>Explored reserves (mill. m$^3$)</th>
<th>Organic sapropel (mill. m$^3$)</th>
<th>Silica sapropel (mill. m$^3$)</th>
<th>Carbonate sapropel (mill. m$^3$)</th>
<th>Mixed sapropel (mill. m$^3$)</th>
<th>% of total explored reserves (mill.m$^3$)</th>
<th>Expected reserves (mill.m$^3$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Brest</td>
<td>92,8</td>
<td>472</td>
<td>26,1</td>
<td>8,9</td>
<td>10,6</td>
<td>5,2</td>
<td>46,0</td>
</tr>
<tr>
<td>Vitebsk</td>
<td>1256,3</td>
<td>220,8</td>
<td>880,3</td>
<td>86,5</td>
<td>77,7</td>
<td>71,9</td>
<td>617,6</td>
</tr>
<tr>
<td>Gomel</td>
<td>87,3</td>
<td>13,0</td>
<td>67,3</td>
<td>3,1</td>
<td>3,9</td>
<td>4,9</td>
<td>1,5</td>
</tr>
<tr>
<td>Grodno</td>
<td>69,7</td>
<td>14,9</td>
<td>18,8</td>
<td>27,2</td>
<td>15,8</td>
<td>3,9</td>
<td>13,3</td>
</tr>
<tr>
<td>Mogilev</td>
<td>22,0</td>
<td>10,4</td>
<td>2,0</td>
<td>6,1</td>
<td>3,5</td>
<td>1,3</td>
<td>1,1</td>
</tr>
<tr>
<td>Minsk</td>
<td>223,8</td>
<td>35,0</td>
<td>85,0</td>
<td>50,4</td>
<td>53,4</td>
<td>12,8</td>
<td>185,3</td>
</tr>
<tr>
<td>Total</td>
<td>1759,1</td>
<td>332,5</td>
<td>1079,5</td>
<td>182,2</td>
<td>164,9</td>
<td>100</td>
<td>868,9</td>
</tr>
</tbody>
</table>

The investigations carried out in the Ukraine and Belarus show high effectiveness of sapropel amendments application on soddy podzolic soils contaminated with radionuclides [10, 11]. The investigations carried out at the Leuven Catholic University (Belgium) show the influence of sapropels of different types from some Byelorussian lakes on decreasing of the content of $^{90}$Sr mobile forms in soil [4].

The main advantage of sapropels as potential amendments consists in a high degree of dispersion of the natural materials, considerable contents of organic matter, both nutrient availability and trace elements.
According to the enormous field investigations inserting of sapropels in soil is essentially (to 5-10 times) reduce radionuclide intake from soil in plants, where the effect is observed during a long period of time.

Using of sapropel amendments is especially perspective on soddy podzolic soils where there are the highest migration coefficients of radionuclides from soil to plants [5, 11].

The natural mineral tripoli is very promising natural material for rehabilitation of agricultural soils contaminated by radionuclides in Belarus due to unique and agrochemical sorption properties. Tripoli is compound polymineral formation that consists of five dispersible, finely agitated phases: opal-cristobalite, X-ray amorphous opal, zeolite, calcite and clay minerals. Because of a high content of calcite tripoli of the deposit "Stalnoye" in Hotimsk district of Mogilev region can be referred to calcareous tripoli, which is characterized by a high enough content of zeolite (up to 30 %).

The important characteristics of physicochemical properties of products, materials and substances obtained on the basis of tripoli is their high adsorptive, catalytic and hydraulic activity, heat-insulating ability, heat resistance and sintering ability. The stores of the mineral tripoli in the republic constitute more than 80 million tons, which are on the territory of Mogilev region, where the considerable part of the agricultural soils is contaminated by radionuclides as well [12].

The application of tripoli and the products on its basis for agricultural production on the soils contaminated by radionuclides will allow considerably increasing sorption ability of soils and moisture capacity of loamy sandy and sandy soils and considerably improving structure of peat bog soils. On the whole that will promote decreasing of radionuclide migration in the agricultural production.

Hydrolyzed lignin is multi-tonnage waste of hydrolyzed production that is of great interest from the point of view of the contents of organic matter and sorption ability in the relation to $^{90}$Sr [7]. Clay-saline slame is potassium production waste that is perspective from the point of view of the contents of clay minerals and high sorption ability in the relation to $^{137}$Cs [8].

4. Conclusions

In conclusion it is necessary to mark that development and application of organomineral amendments on the basis of local natural organic and mineral raw material and industrial wastes will stimulate development of ecologically safe technologies in agriculture of Belarus and organic farming in the territories contaminated by radionuclides. On the whole, realization of the measures mentioned above in Belarus will allow reducing environment load because of decreasing use of mineral fertilizer and, hence, the degree of risk of the population living on the territories contaminated by radionuclides.

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Necessity of the establishment of a national system for the management of radioactive waste in Paraguay

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Abstract. Radioactive wastes derive from the production of nuclear energy and the use of radioactive material in medicine, industry and research. The safe management of radioactive waste is very important for the protection of the human health and the environment. Paraguay does not have nuclear power plants, but the use of radioactive materials in medicine, industry and research generates radioactive waste. One main problem is that the country does not have a national system for the management of radioactive waste. Although radiation sources are under the regulatory control of CNEA, the number of sources in disuse is growing. The increase of radioactive sources in disuse and distributed in different places without proper management becomes unsustainable and urges to have a system to guarantee centralization and security of these sources.

1. Introduction

Paraguay is a country with a surface of 406,754 Km2. Paraguay boarders in the North to Bolivia and Brazil and in the South-West to Argentina. It is part of the Common Market of the South (MERCOSUR), has an approximate population of 6,000,000 million inhabitants, and has two main geographical regions, the Oriental Region and the Western Region named Chaco, with a total of 17 Departments. In the country there are public and private facilities that possess sources in disuse without management. Since most of these facilities are located in the Central Department (Oriental Region), their regulatory control is easier. Table I lists the geographical distribution of sources in disuse without management.

Table I: Geographical distribution of sources in disuse without management

<table>
<thead>
<tr>
<th>Quantity</th>
<th>Source</th>
<th>Half Life</th>
<th>Practices</th>
<th>City</th>
<th>Department</th>
</tr>
</thead>
<tbody>
<tr>
<td>2 (two)</td>
<td>$^{60}$Co</td>
<td>5.27 a</td>
<td>Teletherapy</td>
<td>ASUNCIÓN</td>
<td>CENTRAL</td>
</tr>
<tr>
<td>5 (Five)</td>
<td>$^{137}$Cs</td>
<td>30 a</td>
<td>Brachytherapy</td>
<td>ASUNCIÓN</td>
<td>CENTRAL</td>
</tr>
<tr>
<td>1 (one)</td>
<td>$^{60}$Co</td>
<td>5.27 a</td>
<td>Teletherapy</td>
<td>CAPIATA</td>
<td>CENTRAL</td>
</tr>
<tr>
<td>107 (needles)</td>
<td>$^{137}$Cs</td>
<td>30 a</td>
<td>Brachytherapy</td>
<td>CAPIATA</td>
<td>CENTRAL</td>
</tr>
<tr>
<td>268 (mg)</td>
<td>$^{226}$Ra*</td>
<td>1620 a</td>
<td>Brachytherapy</td>
<td>CAPIATA</td>
<td>CENTRAL</td>
</tr>
<tr>
<td>1 (one)</td>
<td>$^{60}$Co</td>
<td>5.27 a</td>
<td>Gamma Radiography</td>
<td>SAN LORENZO</td>
<td>CENTRAL</td>
</tr>
<tr>
<td>1 (one)</td>
<td>$^{60}$Co</td>
<td>5.27 a</td>
<td>Gamma Radiography</td>
<td>LUQUE</td>
<td>CENTRAL</td>
</tr>
<tr>
<td>5 (Five)</td>
<td>$^{60}$Co</td>
<td>5.27 a</td>
<td>Meter Level</td>
<td>VILLA HAYES</td>
<td>Pte. HAYES</td>
</tr>
<tr>
<td>3 (three)</td>
<td>$^{241}$Am</td>
<td>432 a</td>
<td>Meter Level</td>
<td>ASUNCIÓN</td>
<td>CENTRAL</td>
</tr>
<tr>
<td>2 (two)</td>
<td>$^{241}$Am</td>
<td>432 a</td>
<td>Meter Level</td>
<td>YPANE</td>
<td>CENTRAL</td>
</tr>
<tr>
<td>2 (two)</td>
<td>$^{241}$Am</td>
<td>432 a</td>
<td>Meter Level</td>
<td>NEEMBY</td>
<td>CENTRAL</td>
</tr>
<tr>
<td>2 (two)</td>
<td>$^{90}$Sr</td>
<td>29.1 a</td>
<td>Meter Level</td>
<td>CAPIATA</td>
<td>CENTRAL</td>
</tr>
<tr>
<td>20 (twenty)</td>
<td>$^{241}$Am</td>
<td>432 a</td>
<td>Lightning Rod</td>
<td>SAN LORENZO</td>
<td>CENTRAL</td>
</tr>
</tbody>
</table>

* Conditioned but without storage (IAEA Project)
But there exist medical and industrial facilities that possess and use radioactive material, whose number and geographical distribution is extending. These sources could cause accidents or be misused by terrorists. This is not only a national problem but, considering that the country is part of the MERCOSUR, a problem of regional concern and needs to be adjusted to legal requirements of the region.

2. Objectives

2.1. General objective

A most important general objective for the country is to establish a national system of radioactive waste management, in order to protect the human health and the environment, now and in the future without imposing undue loads to future generations.

2.2. Specific objectives

Specific objectives are to deal with:

- management politics and strategy,
- creation of an effective national juridical frame,
- basic elements of a respective National Management System (infrastructure, adequate human resources and specific equipment),
- clear responsibilities of the regulatory body, the user, the operator of the installation.

3. Principles of the management of radioactive waste

To achieve the objective of safe management of the radioactive waste, the following principle should be fulfilled: *That the management of the radioactive waste will take such a form that an acceptable level is guaranteed.* Issues to be considered are:

- Protection of the human health.
- Protection of the environment.
- Protection outside of national frontiers.
- Protection of future generations.
- Load imposed to future generations.
- National law frame.
- Production and control of radioactive waste.
- Reciprocal dependence between production and management.
- Security of facilities.

4. Conclusion

To fulfil the objective, the country will follow national policies on the management of radioactive waste, in conformity with the enunciated principles, and elaborate strategies that will depend on the circumstances, structures and national priorities as well as on the diverse types of radioactive waste.

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Legislative basis for the foreign spent nuclear fuel management in Russia

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Abstract. In this note, the legislative basis relating to the handling of foreign spent nuclear fuel imported into the Russian Federation for the purposes of the temporary technological storage, or temporary technological storage and further reprocessing, is briefly discussed.

1. Introduction

One of the main objectives of the 'Joint Convention on the Safety of Spent Nuclear Fuel Management and on the Safety of Radioactive Waste Management' is to achieve and maintain a high level of safety world-wide in spent fuel and radioactive waste management, through the enhancement of national measures and international co-operation including, where appropriate, safety-related co-operation.

One of the possible ways for enhancing the safety of the spent nuclear fuel and radioactive waste management could be the use of the existing potential of the countries having the appropriate technologies by means of rendering the service on the spent nuclear fuel and radioactive waste management.

Efforts on the development of the necessary legislative basis and proposals on the modernization of the existing industrial basis relating to the temporary technological storage and reprocessing of the spent nuclear fuel, including the spent fuel imported into the Russian Federation from abroad, were undertaken in Russia.

2. Legislation

The main legislative acts regulating the spent nuclear fuel import into the Russian Federation from abroad are the following:

- Federal Law “On the Use of Atomic Energy”;
- Federal Law “On the Environmental Protection”;
- Federal Law “On the special ecological programmes for the radiation polluted territories rehabilitation”.

According to the up-to-date legislative basis, the spent nuclear fuel import into the Russian Federation should be anticipated (foregone) by the positive conclusion of the respective special commission formed by the President of the Russian Federation.

The state ecological review and other necessary state reviews of the appropriate project related to the spent nuclear fuel import into the Russian Federation from foreign countries for the purpose of temporary technological storage and/or reprocessing should be implemented (conducted) according to the up-to-date legislative basis and the common radiation impact risk reduction, and ecological safety level enhancement, as the result of the project realization (implementation), should be founded as well.

An order of spent nuclear fuel import into the Russian Federation was established according to the main principles of the nuclear weapons non-proliferation, environmental protection, and the economical interests of the Russian Federation, and approved by the Decree of the Government of the Russian Federation named “On the Order for the Import of the Irradiated Nuclear Reactors Assemblies into the Russian Federation”, dated 11 July 2003, No 418.
This Order has the aim to establish the conditions for the broadening of the international co-operation of the Russian Federation in the field of the foreign reactors spent nuclear fuel and products of their reprocessing management, and specifies the order for the nuclear reactors spent fuel assemblies import into the Russian Federation, as well as the order for the returning of the irradiated assemblies or the products of their reprocessing (including the radioactive waste) back to the country from which these irradiated assemblies were imported.

The irradiated assemblies import into the Russian Federation is based on the annual limits approved by the Government of the Russian Federation, taking into account the proposals prepared by the state body governing the use of atomic energy, and co-ordinated with the appropriate regulatory body in the field of the safety of the use of atomic energy, as well as with the local government authorities of the territories where the reprocessing and storage facilities are located.

In particular, according to the Order, the irradiated assemblies import into the Russian Federation is possible only in the cases of positive result of the state ecological review of the “Joint project”, which should be prepared by the designated organizations and should be co-ordinated with the state body governing the use of atomic energy and with the appropriate regulatory body in the field of the safety of the use of atomic energy. The designated organizations should have the necessary licenses to conduct such kind of activity.

2. Joint project

The Joint project is a set of documents, prepared in connection with the forthcoming foreign trade contract pertaining to the irradiated assemblies handling. It should contain, among others:

- the foreign trade contract (with the indication of the resources which are planned to be gained on the basis of the contract realization and the expenses on the handling with the irradiated assemblies and the products of reprocessing);

- special ecological programme (s);

- documents which found the common radiation impact risk reduction and the ecological safety level enhancement as the result of the project realization (implementation), as well as the period of time for the temporary technological storage of the irradiated assemblies and the reprocessing products specified in the contract;

- other documents subjected to the state ecological review in accordance with the existing legislative requirements; as well as the conclusions of state body governing the use of atomic energy and of the appropriate regulatory bodies in the field of the safety of the use of atomic energy.

There are two options specified in the Order for the irradiated fuel assemblies import into the Russian Federation:

- temporary technological storage with subsequent obligatory return back to the country from which these irradiated assemblies were imported;

- temporary technological storage with the further reprocessing.

The conditions of the reprocessing products returning back to the country from which the irradiated assemblies were imported are also specified in this Order.

The organizations authorized to conclude the foreign trade contracts related to the irradiated fuel assemblies import into the Russian Federation were also specified: by the Directive of the Government of the Russian Federation (29 March 2002, No. 380-p).

The document “Statute on the development of the special ecological programmes for the radiation polluted territories rehabilitation” was approved by the Governmental Decree dated 14 July 2000, No.421. These programmes should be included into the Joint project.

The service related to the reprocessing products handling may be rendered to the country from which the irradiated fuel assemblies were imported in case this is in compliance with the principles of the
nuclear weapons non-proliferation. It should be specially indicated in the appropriate international agreements of the Russian Federation.

After the expiration of the period of time for the temporary technological storage, the irradiated fuel assemblies should be returned back to the country from which these assemblies were imported in accordance with the obligations and guarantee of this country.

The amount of the reprocessing products, subjected to be returned to the country from which the irradiated fuel assemblies were imported, should be identified according to the mutually agreed procedures on the basis of the activity equivalence of the imported irradiated fuel assemblies and the activity of the exported reprocessing products taking into account the natural decay of the radionuclides during the temporary technological storage of these irradiated fuel assemblies, reprocessing process and storage of the reprocessing products.

4. Conclusion

The ratification process for the Joint Convention on the Safety of Spent Nuclear Fuel Management and on the Safety of Radioactive waste Management is presently in progress in the Russian Federation. From the legislative and the normative point of view, an adequate basis exists in Russia today for the safe management of the spent nuclear fuel imported into Russia from abroad.
Licensing framework of radioactive waste management in Indonesia

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Abstract. The utilization of nuclear energy has been developed in Indonesia in research, agriculture, health, industry and other fields. Besides the positive aspects, nuclear energy has the potential of radiation hazard. A potential radiation hazard derives from radioactive waste. To control the radioactive waste, BAPETEN has, in the year 2002, established Government Regulation No. 27. This paper addresses present regulation relating to the management of radioactive waste, including the licensing system.

1. Introduction

While nuclear science and technology has been developing, the utilization of nuclear energy in Indonesia has also been developed in the field of research, agriculture, healthy, industry and other fields. Besides having positive aspects, i.e. to increase the people’s welfare and prosperity, nuclear energy has the potential of radiation hazards to the workers, the public and the environment. A potential source of radiation hazards derives from radioactive waste generated from the operations of nuclear installations and radiation facilities. The management of radioactive waste needs to be regulated and controlled to prevent radiation hazards to workers, the publics and to the environment. The control of the radioactive waste management is conducted by the regulatory body (BAPETEN). It is conducted by establishing regulations, carrying out licensing and performing inspections. This paper addresses present regulation relating to the management of radioactive waste, including the licensing system.

2. Review of present regulation

In Indonesia, the highest regulation of the radioactive waste management is the Act No. 10 Year 1997 on Nuclear Energy. The radioactive waste management is governed in one chapter of the Act (Chapter IV) and stipulated in 6 articles (articles 22 – 27) [1]. In these articles, some provisions are established such as:

a. Classification of radioactive waste;

b. Performing the radioactive waste management by executing body;

c. Obligation of radioactive waste generator;

d. Final repository;

e. Transportation and storage of radioactive waste.

The implementation regulation of the Act No. 10/1997 on Nuclear Energy is Government Regulation No. 27 year 2002 on radioactive waste management. The objective of radioactive waste management is to protect the safety and healthy of worker, public and the environment from radiation hazards and or contamination.

Based on GR No. 27/2002, the regulations of radioactive waste include:

- classification of radioactive waste,
- licensing management,
- process, transportation and storage of radioactive waste,
- quality assurance programme,
- management and monitoring of the environment,
- process of explorations and exploitations of nuclear and other ores,
- decommissioning programme.

In addition, a decree of the chairman of BAPETEN has been established: DCB No. 03 of the year 1999 on provision of radioactive waste safety.

According to Chapter IV of Government Regulation No. 27, 2002, the licensing process on radioactive waste management is as follows [2]:

- each person or body performing the nuclear energy utilization should state to the regulatory body that the radioactive waste will be re-exported to an original country or submitted to the executing body to be managed. The executing body managing the radioactive waste in Indonesia is BATAN – Centre for Development of Radioactive Waste Management (CDRWM). The re-export of radioactive waste to the original country should obtain the approval of regulatory body;
- the proof of the performance of re-export to the original country should be submitted to the regulatory body;
- in case the radioactive waste will be managed by the executing body, the regulatory body will inform the executing body;
- the executing body performing the radioactive waste management should obtain the license from the regulatory body;
- the construction and operation of the facility of collecting, classifying or processing or temporary storing radioactive waste arising from nuclear material exploration should obtain the license from the regulatory body;
- the construction and operation of final repository for radioactive waste should obtain the license from the regulatory body. The licensing steps involve siting, construction and operation licensing.

3. Overview on licensing of radioactive waste management

According to the Act No. 10/1997 and GR No. 64/2000 on licensing of nuclear energy utilization, any person or institution using nuclear energy should have a license issued by the regulatory body. The license will be issued after the radioactive waste management fulfils the safety requirements. The requirements for obtaining the license according to GR No. 64/2000 for a facility of radioactive waste management are as follows:

A. having industrial license or other related government institution license,
B. having required facilities,
C. having the qualified personnel,
D. having working procedures,
E. having radiation protection equipment and tools,
F. submitting safety analysis report,
G. submitting analysis of environment impact,
H. submitting construction requirement.

In performing a review of the licensing requirements, BAPETEN uses government regulations and provisions on nuclear and radiation safety such as:

- GR No. 64/2000 on licensing of nuclear energy utilization,
- DCB No. 01/1999 on radiation safety,
- GR No. 27/2002 on radioactive waste management,
- DCB No. 03/1999 on provision of radioactive waste safety.
CDRWM-BATAN has proposed licensing with enclosing licensing documents to BAPETEN (at that time the Atomic Energy Control Bureau –BATAN). BAPETEN carried out the safety evaluation based on the regulation above. The licensing documents submitted include:

1. the safety analysis report;
2. the quality assurance programme;
3. some procedures (i.e. decontamination procedure, environmental monitoring and management procedure, radioactive monitoring procedure, etc.);
4. the emergency preparedness programme.

The Safety Analysis Report (PSAR) has been proposed by CDRWM and has been revised many times according to the recommendations given by BAPETEN through safety evaluation. BAPETEN performed inspections and QA audits to assure that the safety requirements were met during the radioactive waste management activities. BAPETEN also has issued the re-export licensing to radioactive waste generator having the statement that the radioactive waste will be returned to the original country.

4. Discussion and conclusion

In Indonesia, the executing body of radioactive waste management is CDRWM- BATAN. BAPETEN does not yet have a very specific safety review plan on radioactive waste management. It is up to the regulatory body to evaluate or assess the submitted safety analysis report. In issuing regulations, respective guidance and the safety review plan on radioactive waste management, BAPETEN had gained considerable experience. BAPETEN has evaluated the Safety Analysis Report proposed by CDRWM and carried out inspections for checking the safety of the radioactive waste management activities.

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International advisory group for Olkiluoto investigations INAGO

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Abstract. The nuclear waste management company Posiva Oy is building an underground rock characterisation facility ONKALO at Olkiluoto, Finland, which should provide important information for the main research, development and technical design areas for the planned spent fuel repository. As a part of the QC/QA activities Posiva Oy has established an international advisory group to support its management in making efforts towards the company’s strategic goal of submitting a successful application for the construction license of the spent fuel repository by the end of 2012. The group focuses on the work in the ONKALO, but may also take a broader view of other site investigations activities at Olkiluoto. The objectives, working methods and first results of this group are presented.

1. Background

On the 18th of May 2001 the Finnish Parliament ratified the Decision-in-Principle (DiP) taken by the Government that “[construction of the disposal facility for spent nuclear fuel]… at Olkiluoto in the municipality of Eurajoki is in the overall interest of society”.

As stated in the report of the Parliament’s Finance Committee, the DiP means that the disposal project can now proceed to the construction of an underground investigation facility and more detailed, site-specific studies. The need for underground rock characterisation is also expressed in the YVL Guide 8.4 by the Radiation and Nuclear Safety Authority (STUK), listing the construction and operation of an underground research facility as the next main milestone after the selection of the disposal site [1]. The Decision-in-Principle is valid for 15 years from the date of the Parliament’s ratification, which means that the application for the construction licence has to be submitted to the Government during this period.

1.1. ONKALO\(^1\) underground rock characterisation facility

In January 2001 Posiva published a programme for research, development and technical design work (RDD) in the pre-construction period [2]. The programme laid special emphasis on the ONKALO underground rock characterisation facility, which should provide important information for the main RDD areas. The information and experience from the ONKALO investigations, experiments and construction work was seen as key input for the Preliminary Safety Assessment and the application for the construction licence for the disposal facility, which will be submitted in the early 2010’s.

The ONKALO UCRP is based on the same goals and considerations as the RDD programme but gives a more detailed description of the planned underground work. According to the RDD programme, the underground investigations are particularly designed:

- to enhance the current geo-scientific model of the site by a more detailed description of the explored volume of the rock mass;
- to collect data that cannot be obtained from surface-based boreholes;
- to test interactions between the rock and the engineered barrier system.

\(^1\)ONKALO is an acronym based on the Finnish language expression for “Olkiluoto Rock Characterisation for Final Disposal”. The word “onkalo” also means “cave” (in Finnish).

A particular purpose of the underground investigation programme is to assess whether the present conclusions on the suitability of the Olkiluoto site for the spent fuel repository can be confirmed by investigations carried out at the actual depth of the repository.
1.2. **International experience**

Experience on constructing underground research laboratories (URLs) and on conducting underground experiments has been gained in several countries during the past few decades. In recent years Posiva has participated in the R&D work of several of these facilities and utilised this experience in the planning of the ONKALO facility and the related underground research and characterisation programme.

The development of the ONKALO programme has partly been based on the experience from other URLs. However, all currently existing underground laboratories in crystalline rock are of the generic type, i.e. they are not located on the actual disposal sites and have been designed for generic studies only. Compared to these facilities the underground research programme of ONKALO is to a greater extent focused on the characterisation of the site-specific properties of the rock mass, and specific underground experiments are mainly related to tests and experiments in which the site-specific conditions at Olkiluoto have to be taken into account. In addition, the plan to use the ONKALO later as a part of the actual repository means that the construction and operation of the ONKALO should comply with the QA and other requirements set for nuclear facilities.

2. **International advisory group INAGO**

With the start of the ONKALO excavation Posiva took an important step towards the implementation of the plans for disposal of spent fuel in the Olkiluoto bedrock. The ONKALO should make it possible to assess whether the previous assumptions about the rock characteristics at the repository depth hold with the reality and pave the way for the application for the construction license. Moreover, if the assumptions about the suitability of the Olkiluoto site for the repository can be confirmed, the Onkalo is likely to become a part of the repository and, therefore, any work at ONKALO will have a direct link to the implementation of the disposal facility itself.

Recognising the needs for high quality and expertise Posiva is also stepping up its QC/QA activities. As a part of this Posiva decided to establish an international advisory group to support Posiva’s management in making decisions and efforts towards reaching the company’s strategic goal of submitting a successful application for the construction license of the spent fuel repository by the end of 2012. The group focuses on the work in the ONKALO, but may also take a broader view of other site investigations activities at Olkiluoto. The practical work basically includes reviews of programmes and important results and in-depth discussions with the responsible investigators and managers.

2.1. **Objectives**

In more detail, the objectives of the group are to

- advise Posiva on issues related to the performance and objectives of the underground investigations programme and the management of the underground activities, giving also attention to the fact that the ONKALO activities should not unduly compromise the site characteristics of importance for long-term safety.
- advise Posiva on the integration of the ONKALO investigations with the overall characterisation of the Olkiluoto site and other main activities of Posiva.
- review main results and reports from the activities in ONKALO and on the Olkiluoto site in general.
- assess the scientific level and status of the research and investigations from the international perspective by bringing in the experience and knowledge obtained in other research programmes with similar interests and contexts.

In practice, this means reviewing

- the relevance of the investigations performed or planned for various purposes of the repository development programme (level of site understanding, safety case, design of the repository);
- the appropriateness of the methods planned or used in data acquisition, interpretation and modelling for the intended purpose (quality, scientific level);
• Posiva’s competence and capacity for the work planned;
• the usefulness of the results achieved for their intended purpose.

2.2. Members

The INAGO has 5 members representing different kinds of known expertise in geosciences. The idea is to collect experience from ongoing site investigations programmes and earlier rock laboratory projects that have relevance for the work at Olkiluoto. The members do not have any direct involvement in Posiva’s current investigations activities, nor do they have any other vested interests in Posiva’s ongoing programme. The members are prepared as a minimum for two regular meetings annually, some additional review capacity is appreciated in between when possible. In addition the INAGO has a chairman (the author) and a scientific secretary.

2.3. Mode of operation

The main activities of the INAGO can be described as follows:

• to review goals, contents, methods and practical approaches presented in the investigation programme for ONKALO and to suggest improvements or reconsiderations. The overall programme for the ONKALO investigations is explained in the URCP report [3]; more detailed information is given in the TKS-2003 programme [4] and the future three-year programmes to come.
• to review results and achievements as presented in oral and written reports and by the programme managers, principal co-ordinators and investigators, and advise on possible improvements.

Regular review meetings are normally held at most twice a year, each one lasting 2 – 3 days. These meetings sometimes focus on a comprehensive overview of the whole programme, sometimes on a specific theme. Typically, a review meeting includes presentations, discussion within the review group and the preparation of the review statements. In addition, the group members may review on their own choice specific documents or reports.

2.4. Work done

The work of the INAGO group was started in spring 2004. So far there has been two review meetings, INAGO-1 in June 2004, and INAGO-2 in December 2004, and the third meeting is scheduled for June 2005.

The first meeting, INAGO-1, concentrated in the evaluation of the future programmes for the Olkiluoto investigations, the Underground Rock Characterisation and Research Programme UCRP [3], the programme for monitoring [5] and the three-year programme TKS-2003 [4] (site investigation parts). Another point in the agenda was the planning of further work programme of INAGO.

The second meeting, INAGO-2, started with the developments made in ONKALO investigations, and concentrated then in the evaluation of the ONKALO quality assurance programme and of the STUK review of TKS-2003.

3. Conclusions

The INAGO way of working seems to have several advantages compared with ordinary reviewing process. The INAGO group is independent of both the nuclear fuel waste repository company and of official controlling organisations, and has the scientific freedom to make recommendations to the investigation programmes.

All of its members have previous experience in nuclear fuel waste repository investigations in their home countries. They represent different branches of science covering all the main investigation programmes in ONKALO. This gives an additional synergy advantage on a single investigator or research group, which in some cases may become blind to everything else than their speciality.

The first active year of INAGO has been very satisfactory to its members, and assisted the ONKALO team to make early adjustments to its investigation programmes for an easier reach of the final goal, the construction licence for the disposal facility.
REFERENCES


Strategies for technical confidence building in the NUMO HLW disposal programme

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Abstract. Building confidence in the safety of geological disposal is a challenge faced by all waste management programmes. It is recognized that confidence building measures are needed for interacting with the technical community and also with the general public. Although both aspects are critical and strongly overlap, this paper addresses primarily the strategies for enhancing technical confidence. This is an absolute pre-requisite for engaging a wider public. Unless and until the scientific and technical community actually engaged in implementing repositories has great confidence in their feasibility and safety – and can successfully and convincingly communicate this in the wider scientific circles – then it is premature to take the debate into the public arena. The Japanese HLW disposal organization, NUMO was created only relatively recently (2000). It instantly faced the challenge of initiating one of the largest HLW disposal programmes in the world – and it has chosen to tackle the key challenge of identifying potential repository sites by using a completely transparent volunteering approach. This paper describes the measures that NUMO has taken to establish its scientific credibility at home and abroad.

1. Introduction

Building confidence in the safety of geological disposal is a challenge faced by all waste management programmes. The argumentation to be presented is extremely interdisciplinary and the audiences to be addressed are very diverse. It is recognized that confidence building measures are needed for interacting with the technical community and also with the general public. Although both aspects are critical and strongly overlap, this paper addresses primarily the strategies for enhancing technical confidence. This is an absolute pre-requisite for engaging a wider public. The scientists and technicians actually engaged in implementing repositories must themselves have great confidence in their feasibility and safety. When this has been achieved, then they can more successfully and convincingly communicate the facts to the wider scientific community. These communications between scientists can efficiently use the common language of science, for example allowing a rational discussion on residual uncertainties, the impossibility of zero-risk activities etc. The resulting "internal, scientific debate" is a valuable or perhaps even a necessary preparation for wider public discussions. Although the dialogue with the public needs to be structured and led by specialists in the field of communication, the direct involvement of members of the scientific staff can greatly enhance the credibility of the waste management organization – provided that they have been properly prepared for this task.

This paper describes the measures that the Japanese HLW waste management organization, NUMO, has taken since its establishment in 2000 to establish and maintain technical confidence in its capabilities to achieve the ambitious goals it faces.

2. NUMO mission, structure and programme

NUMO was set up in 2000 after several years of preparatory work by the Japanese High Level Radioactive Waste Project and by the national laboratories [1]. The mission is to site and implement a repository for all of the vitrified HLW that will result from reprocessing of Japanese reactor fuels. NUMO has been set up as a project management organization charged with fulfilling the responsibilities of the Japanese producers of HLW, with government control being exercised primarily through establishment of a national policy and reliance on independent oversight by regulators. This
type of structure has been shown internationally to result in implementer organizations that can work efficiently towards repository realisation.

In the technical area, the two primary units at NUMO are the groups responsible for repository engineering design and safety [2] and for siting strategy [3]. As outlined below both groups have developed strong technical programmes, established active working groups, made extensive use of international expertise and gone to great lengths to make their work open and transparent.

3. NUMO's siting strategy requires technical and public confidence

Governmental policy in Japan determined that the site selection for a HLW repository should be a phased process. The schedule for the multi-year programme to develop the repository foresees the start of operations only in the period 2033-2037. The process begins with the identification of Preliminary Investigation Areas (PIAs), from which Detailed Investigation Areas (DIAs) are selected before moving to full characterisation of a preferred site. The Law in Japan also specifies that NUMO must work transparently in all its activities. On this basis, and having observed that failures in siting programmes world-wide are more often due to societal problems than to technical issues NUMO has chosen an “open solicitation” approach for finding candidate sites. NUMO has invited municipalities throughout the country to consider volunteering as candidate areas for exploring the feasibility of hosting a final repository for HLW [4,5]. This open approach is extremely dependent on NUMO's ability to establish sufficient trust and confidence in the organization.

The reason is that communities are expected to volunteer before any specific studies of their geological suitability have been made. This implies the community must have confidence that NUMO will be willing to break off investigations if negative technical results are produced in any subsequent stage of investigation. How has NUMO aimed at establishing trust that it will act in this responsible fashion? Before requesting offers, two years were spent preparing a comprehensive set of documents for inclusion in an information package that was then sent to over 3000 municipalities [4]. The technical documents described the engineering and safety concepts behind geological disposal and also laid out the geological and other siting factors that would be used to evaluate the potential suitability of volunteer sites. Further documents detailed the economic effects on communities that could result from their willingness to perform a national service by hosting a repository. The high quality and transparency of the solicitation documents was assured through careful preparation and review, involving all NUMO technical staff as well as chosen external experts. A key role was played by the International Technical Advisory Committee (ITAC) and the Domestic Technical Advisory Committee (DTAC) that NUMO established at the beginning of its work.

4. NUMO advisory bodies

Recognising the benefits to be gained by integrating national and international experience into a rapidly growing programme, NUMO already in June 2001 set up a Domestic and an International Technical Advisory Committee. DTAC was formed with most members coming from Japanese academic institutions. DTAC operates with two Japanese expert Subcommittees headed up by members of the parent committee, one on the Geological Environment and the other on Engineering Technology and Performance Assessment. The former helped develop a consensus view of Japanese and international experts on the long-term stability of tectonic plate systems in the Japanese environment and led to the definition of siting factors to be examined in the selection process for PIAs. The range of applicable engineering technologies and framework for performance assessment were also discussed in DTAC and improvement of procedures governing the flow of basic information needed to select the design concepts for specific volunteer sites is one of the key developments. The output of DTAC has been reflected in the reports included in the NUMO information package [4].

In NUMO’s case, the potential benefits of making use of advisory groups are particularly high for two main reasons. Firstly, the Japanese HLW disposal programme is moving into an active phase at a time when a number of other national programmes have completed milestone projects from which much experience can be gained. Secondly, the NUMO open volunteering process implies a) that the communities must have confidence in the technical capabilities of NUMO, as well as in its openness
and honesty and b) that a very wide range of geological settings could require to be characterized and matched to suitable, safe repository design concepts.

In order to address these challenges, ITAC was constituted by extending invitations to individuals very familiar with the disposal programmes of Canada, Finland, France, Germany, Sweden, Switzerland, UK and the USA. These represent countries with a wide variety of geological environments and a diversity of implementation structures. Up to December 2004, eight meetings have taken place in Tokyo. Much of the early work was devoted to helping prepare the extensive technical documentation issued by NUMO when it began soliciting volunteer sites. After the start of open solicitation, ITAC supported the compilation of more detailed technical documents published in 2004 [6,7]. In addition, ITAC meetings have included special technical sessions comparing and contrasting worldwide national approaches to key disposal issues, such as site characterisation, quality management systems and treatment of time scales in performance assessments.

Equally important for NUMO is the need to make the DTAC and ITAC work open and accessible to the public. A highly successful joint public meeting of ITAC and DTAC, was organized in Tokyo in December 2003. In mid 2004, an open forum discussion including talks from NUMO and ITAC, as well as a moderated panel discussion attracted 500 participants. These open meetings including questions and discussions are also documented on the NUMO web site <www.numo.or.jp>. ITAC and DTAC function, both as a vehicle for providing “hands on” technical advice and also as an important component of NUMO’s on-going efforts to enhance public confidence in the safety of deep geological disposal of radioactive wastes [8].

At a more detailed technical level NUMO has also taken pains to ensure that it has the best possible technical support. A prominent example is the International Tectonics Meeting (ITM), organized to tackle scientific challenges that Japan faces more directly than other programmes. These are the potential effects of volcanic activity and of rock deformation on repositories implemented in the relatively active geological environment of Japan. The ITM group includes foreign specialists from the New Zealand, UK, the USA and it interacts strongly with the Japanese geological community. The work of the group is leading to a greatly enhanced understanding of the phenomena mentioned and its results are being published in leading journals [9], thereby counteracting the often stated superficial view that Japan as a whole is geologically unsuited for deep disposal.

A further technical area in which NUMO is – with support from external advisors - moving to the forefront of international research and development concerns the widening of the spectrum of repository design concepts that are being considered. This is in part necessary in Japan because the very diverse geology, together with the open volunteering approach, means that NUMO engineers and safety analysts must be prepared at any time to consider whether a safe repository concept can be developed for any specific potential site that might be volunteered. More work on engineering concepts is, however, also justified by the fact that the original repository concepts developed in many countries, e.g. Sweden, Finland, Canada, Germany and Switzerland were intended more to demonstrate basic feasibility than to be optimized designs for implementation.

5. **NUMO international collaboration and public activities**

In addition to the specific use of technical advisory groups, NUMO has also maximized the benefits it can obtain from international collaboration in a number of other ways [10]. Bilateral information exchange agreements have been signed with Posiva, Nagra, SKB, ANDRA, U.S.DOE and Nirex. NUMO became a member of EDRAM (International Association for Environmentally Safe Disposal of Radioactive Materials) in May 2001. Joint projects on subjects such as those mentioned above, and also collaborative studies on topics such as an international workshop on bentonite-cement interaction [11], international databases [12] have been initiated. As such, they record the technical development of NUMO’s programme and complement materials presented by NUMO staff at many national and international conferences, symposia and workshops and published in technical journals (e.g. [1,5]). An important step was to delegate staff to participate in active foreign programmes. The key technical lines at NUMO – repository design, safety and siting strategy – were headed in their build-up phase by scientists who had been fully integrated into active disposal programmes abroad before taking up their NUMO duties.
To promote public understanding and to encourage application, it is essential to initiate and develop nation-wide discussion on HLW issues. This requires sufficient understanding of the characteristics of HLW and disposal options. NUMO organized information meetings at 31 different locations out of a total 47 prefectures during the period between December 2001 and November 2002, with a total of approximately 5,000 participants from the public. Since June 2003, NUMO and local newspapers have jointly hosted round-table talks with local opinions-leaders at 32 locations. Information campaigns have been conducted in leading newspapers, on TV, and in magazines. The information package, technical documents, booklets, videos and pamphlets, some of which have been produced in English, are available from NUMO web site.

6. Conclusions

Building confidence in the feasibility of implementing safe deep geological repositories is a challenge facing all HLW disposal programmes. Successfully achieving this objective in broad public and political circles is certainly a bigger task than doing so in the technical community. NUMO is aware of this and actively monitors public attitudes. However, sufficient confidence in the technical community is a necessary (though not sufficient) condition for wider trust. NUMO has taken specific steps to establish that it itself is a credible organization, that it manages its work using the best advice from external experts and that it has a transparent long-term programme against which its progress can be judged. The result is that the Japanese HLW disposal programme today is state-of-the-art and is also one of the most active and innovative national programmes.

REFERENCES

Implementation of a requirements management system for the Japanese HLW programme

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Abstract. The Nuclear Waste Management Organization of Japan (NUMO) is presently establishing a Requirements Management System (RMS) which has two main goals: providing an overview of the complex balancing of technical and socio-political requirements which form the basis for decision making in the nuclear waste management field; serving as a tool to guide day-to-day management of long-term, multi-disciplinary projects. The complex interactions between different requirements and the way in which these change as the programme develops makes RMS implementation a challenging task, but this should be aided by a relational database software tool which is currently under development.

1. Background

Although the terminology is by no means standardised, Requirements Management (RM) can be defined here as the process by which:

- the fundamental needs and constraints ("requirements") in a programme are documented in a structured manner to facilitate their consistent use as a basis for decision making,
- proposed or actual changes in requirements (especially associated with the synthesis at the end of major project stages) are recorded and their consequences assessed systematically,
- project managers can see how their work and their decisions affect decision making within the wider project context,
- project staff are informed of new or modified decisions that may affect their work and the decisions they take,
- a historical record of project decisions and the factors that influenced them is maintained.

In principle, this is a component of conventional project or programme management. Increasingly, however, RM is seen to be a topic requiring special consideration in large or complex engineering projects. In national nuclear waste management programmes, there has been a recent trend towards formalisation of this process in a RMS, due to the difficulties arising from the very long timescales involved (some already entering their 4\textsuperscript{th} decade) and rapidly expanding complexity as projects move from desk studies towards implementation [1]. Such formalisation is also consistent with the latest Quality Management guidelines (ISO 9001–2000), which emphasise rigorous process description / documentation.

For NUMO, development of a RMS has been assigned high priority because of the special boundary conditions set by the Japanese HLW programme [2], its commitment to openness and transparency and its very limited resources of staff with long-term, multi-disciplinary expertise in this area. Some of this work is carried out in collaboration with NUMO's Swiss partner Nagra.

2. The NUMO concept for a requirement management system

NUMO recognises that there are a range of requirements which constrain development of a Japanese HLW project. These can be classified as falling into hierarchical levels, e.g.:
primary: these would be absolute and, ideally, unambiguous. These would generally be externally imposed upon NUMO and would include laws, regulations, etc.;

secondary: these are NUMO commitments which include, for example, local acceptance & involvement, openness & transparency, etc.;

tertiary: these are pragmatic or programmatic constraints which are necessary for concretisation of the project – for example the assumed inventory for the first repository and the cost ceiling for this project;

quaternary: these include all other derived "working level" requirements, which result from one or more upper level constraints. This would include parameter values such as minimum canister thickness, minimum and maximum bentonite density, etc.

It is clear that such requirements are not fixed for the entire duration of a repository programme (in the order of a century) and, indeed, many of the primary requirements have not yet been finalised for the Japanese case. Nevertheless, the higher level requirements may be expected to change less with time than those in lower levels. The derived "quaternary" requirements, in particular, may change considerably during the process of site selection and characterisation and could include simple "place holders" which can be specified in detail only at later project stages.

Requirements provide the input needed to produce the specifications which will be used to plan and build a repository. These specifications describe:

- site selection and characterisation programmes;
- repository concepts and implementation programmes;
- engineering technologies for construction, operation and closure
- knowledge base: models, databases, computer codes (including archiving & communication);
- supporting methodologies for assessing: safety (operational and post-closure); quality; resource needs (manpower, R&D, material, financial); environmental impact;
- deliverables for the licensing process.

Decisions arise where developing such specifications involves choices between alternatives. Requirements provide the input required to make these decisions, but this may not be a simple process when many different requirements need to be addressed for a particular decision and, further, such requirements may be:

- qualitative or quantitative, e.g.: site must be geologically stable; depth must be greater than 300 m;
- consistent or contradictory, e.g.: ease of handling – overpack should be as light as possible; reduce risk of sinking: overpack density should be as low as possible; reduce surface radiation dose: overpack should be as thick as possible;
- well defined or still open, e.g.: operational safety must be guaranteed; a long institutional control period may be required.

The RMS does not, therefore, in any way replace the expert judgement necessary to reach complex decisions – but rather helps to ensure that all relevant requirements are considered in this process and the justification for the decision is adequately documented. Especially in the case where contradictory requirements have to be balanced, the weightings assigned to each need to be clearly justified as such weightings may also change considerable as the programme evolves and moves closer to implementation at a specific site.

3. Steps towards requirement management system development and implementation

The overall aim of the RMS is thus openness and transparency in decision-making. However, this overarching aim can be considered in terms of subsidiary aims that impact on day-to-day running of a
The first of these is record-keeping - providing a programme overview as a structured record of all key programme decisions to date and the factors that influenced them (especially important for inexperienced staff and as input for licensing). A more challenging application is in programme planning:

- to anticipate future decisions, the key factors that will influence them, and the requirements that these decisions will impose on major programme activities,
- to explore consequences (in terms of the decisions affected) of alternative requirements on system components, and alternative decisions and boundary conditions,
- to focus R&D by identifying needs and setting priorities.

Associated to this is a role in programme management, ensuring clear communication between programme groups (current requirements, consistency of decisions, databases etc.). This should better integrate decision-making, so that all critical interactions are noted (and acted upon where necessary) and hence programme risks are reduced.

In order to test how this could be implemented in practice, as a first exercise the system of engineered barriers developed in the H12 project [3] were analysed in terms of flow of requirements leading to the decisions relating to the technical specifications for these barriers. According to the structured approach for repository design [2], the range of top-level requirements considered is comprehensive, including legal (waste type, geological stability, repository depth), programmatic (safety concept based on H12 barrier system, implementation schedule, safety (operational and post-closure), engineering feasibility / practicality, acceptance (political, social, financial, academic) and environmental (SEA and EIA) constraints.

Even for these rather simple barriers, representation of the complex interaction of requirements for a specific barrier property, between different properties of a particular barrier and between barriers is very difficult to present in 2 dimensions. The derived flow charts could be summarised as checklists for individual barriers (e.g. Fig. 1) or flow charts showing how the main requirements are expected to change with time, particularly related to key programme milestones. Nevertheless, handling such problems is fairly standard using relational database approaches and hence it was decided to proceed with development of a suitable software tool to manage the resulting study output.

In general, the resulting RMS tool should:

- take full account of the interconnected nature of decisions and requirements – the outcome of one decision is often a factor influencing other decisions,
- be flexible enough to allow for the changing nature of requirements and the possibility of revised decisions as a project progresses,
- accept and record these changes in a controlled, orderly, well-structured manner, supervised, for example, by a system administrator,
- incorporate the concept of "ownership" – each user is authorised to propose modifications (e.g. via the system administrator) to a limited subset of requirements and decisions,
- allow feedback from users affected by any decision/information changes,
- provide adequate security – allowing limited access to sensitive information,
- avoid laborious upgrading procedures and the risk that different versions are running simultaneously,
- be centrally hosted so that different versions are not running in parallel.

Apart from the last 2 points, which relate to specific software and how it is run, the key point about the RMS with respect to its users is that it is structured to give any user the appropriate level of detail in areas relevant to his/her responsibilities, whether the user is a senior manager needing a strategic overview or contemplating the repercussions of new legislation, or a technical project manager needing to update decisions based on new information.
FIG. 1: Derived checklist for EBS component vitrified HLW

4. Conclusions and a look into the future

The formal mapping of requirements onto the H12 EBS specifications has already been a useful exercise for NUMO, indicating the potential impacts of the very different boundary conditions for H12 (generic feasibility study focusing on post-closure performance) and for a repository project which has to be implemented safely and cost-effectively in an actual volunteered site. This will be extended further to consider the entire repository (EBS plus host rock, underground infrastructure, surface facilities, etc. - including variants from H12). In close collaboration with a software development team, it is planned to then develop a RMS tool which will be useful both within NUMO for programme and project management and externally to facilitate open and transparent communication and, finally, licensing.

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A structured approach for stepwise design of HLW repositories tailored to volunteer sites

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Abstract. The Nuclear Waste Management Organization of Japan (NUMO) has initiated a volunteering approach to siting a repository for HLW. This places special constraints on the process of developing repository designs; in particular, the need for a high degree of flexibility in order to respond to the specific geological environments which may come forward. In order to meet this challenge, NUMO has developed a highly structured process for tailoring designs to siting environments, which comprises: establishment of a catalogue of design components which can be combined as required to meet site properties, a bottom-up approach to assess the limitations of specific repository components within the range of geological characteristics expected in Japanese sites, top-down use of multi-attribute analysis to compare sites and designs, iterative feedback from design studies to site characterisation and R&D plans, an integrated requirements management system; ensuring transparent documentation of all decisions and assisting co-ordination of project development. The staged site-selection procedure provides hard deadlines for input from the repository design team for deciding: on the basis of literature studies, which volunteers will be carried forward as “Preliminary Investigation Areas” and subject to surface-based field work, which of these will then be nominated as “Detailed Investigation Areas” and further characterised with an underground test facility, and the final choice of the repository site. As the first two of these deadlines should fall within the next decade, NUMO has leapfrogged the more traditional analysis of idealised repository concepts to concentrate on designs which would be practical and safe to construct and operate and on the “next generation” PA tools required to analyse these on a site-specific basis.

1. Background

Based on the "Specified Radioactive Waste Final Disposal Act", the Nuclear Waste Management Organization of Japan (NUMO) was established in 2000 with the remit to develop a project for the disposal of vitrified HLW resulting from the reprocessing of power reactor fuel. This act is prescriptive in specifying deep geological disposal in Japan at depths greater than 300m below surface and a phased process of siting. NUMO initiated siting with a call for volunteers which was sent to all 3239 municipalities in Japan. The call for volunteers includes clear exclusion criteria to avoid areas which would be unsuitable in terms of geological stability or conflict with utilisation of natural resources (English translations of the documents used for this call for volunteers are available on http://www.numo.or.jp/english/index.html).

The wide range of potential siting environments which may result from this process poses a particular challenge for the development of repository concepts. Apart from the need to maintain a very high degree of flexibility in order to assure compatibility with whatever geological environments are encountered in the volunteers, the job is made even more difficult by limited resources of experienced staff and a very ambitious timescale for implementation. In order to respond to these challenges, NUMO has utilised the time until volunteers come forward to develop a highly structured approach to repository concept development. The repository design thus develops in a staged manner – becoming more detailed as siting environments become better defined. As the repository concept both requires input from, and provides guidance for, the site characterisation work and associated R&D, these efforts are closely co-ordinated.
2. **The structured approach**

Repository concept development was identified as a priority work area at an early stage and has been the topic of collaboration with NUMO’s Swiss partner Nagra since 2001. A “repository concept” is defined [1] as a conceptual design of all surface and underground repository structures, along with a description of how the repository can be constructed, operated and sealed. This also includes an evaluation of operational and long-term safety and an assessment of costs and socio-political impacts. The concept is dynamic, evolving with the programme as it moves from early generic studies through to siting and, eventually, licensing for construction and operation. Indeed, continual evolution during the operational period is also expected, as experience is gained and technology develops.

The strategy for iterative development of the repository concept as the process of site selection and characterisation progresses is illustrated in Figure 1. The starting point is provided by over two decades of Japanese R&D on HLW disposal carried out by a range of organisations, led by JNC, before NUMO was established. This work, most recently summarised in the H12 reports [2], involved a generic evaluation of the requirements for a safe repository in the types of rocks and siting environments expected to be found in Japan. The basic repository design developed by JNC involved sealing vitrified waste in a thick steel container (the “overpack”), which is surrounded by a highly compacted bentonite layer (the “buffer”). A number of variants of the emplacement layout and the materials involved were examined in the H12 study, but there was no attempt at optimisation to improve operational practicality or to match the exact conditions to be expected in site-specific host-rock environments.

Since NUMO was established, these foundations have been built upon in order to develop a catalogue of repository components which could be assembled to produce design options appropriate to particular sites [1]. This was carried out in a “bottom-up” manner by considering the constraints set by the range of mechanical, hydrological, chemical and thermal properties expected at potential sites on the performance of these individual components. Note that, although long-term safety is an essential performance requirement, a much wider set of "design factors" are considered in this assessment:

- Long-term safety: robustness of the post-closure safety case;
- Operational safety: conventional and radiological safety of construction, operation and decommissioning;
- Engineering feasibility: fundamental feasibility of construction and operation to defined quality levels;
- Engineering reliability: practicality of implementation in view of boundary conditions (e.g. emplacement rate) and robustness with regard to operational perturbations;
- Site characterisation / monitoring requirements: effort required to satisfy technical requirements for site characterisation and monitoring data;
- Retrievability: ease of retrieval after emplacement;
- Environmental impact: extent of all environmental impacts associated with repository implementation;
- Socio-political and economic aspects: factors contributing to costs and acceptance by all key stakeholders.

In the process of refining the design, engineering and performance assessment constraints also need to be considered, especially as these may interact in a complex manner. This is best illustrated by an example (Fig. 2), considering the way in which the basic design could be modified to respond to a volunteer site with limited area, which requires layouts with small footprints to be examined. An inevitable consequence of higher emplacement density, however, is higher thermal loading which could result in temperatures in the buffer exceeding the 100°C limit set in H12. Even without changing the fundamental EBS components (HLW – steel – bentonite), a number of ways have been identified in which the design or institutional procedures could be modified to ensure that performance is not compromised. Nevertheless, as indicated in Figs. 1 & 2, these options may give rise to requirements for further R&D or specific site characterisation activities.
FIG. 1: Illustration of the iterative tailoring of reference repository concepts to site characteristics.

For any site, a number of different designs may need to be compared and, at various stages of site selection, the pros and cons of different sites need to be assessed. Such comparisons need, in principle, to take aspects of all the listed design factors into account and a hence a “top-down” Multi-Attribute Analysis (MAA) approach has been developed for this purpose.

FIG. 2: Illustration of the wide range of possible organisational and design responses to the case of a site requiring a compact repository layout.
Even without a site, such MAA exercises have proven invaluable for improving the communication between NUMO’s site characterisation, repository design, safety assessment and public communication groups. Such exercises have also become increasingly realistic, the last comparing repository construction and operational designs produced by teams from NUMO, Nagra and Nirex.

3. Moving towards DIA selection

As information presently accumulated may be used to support important future decisions or in license applications, the importance of a well established Quality Management System (QMS) has been recognised by NUMO. Even at the stage of literature survey of volunteers, procedures have already been defined for reviewing and classifying data according to its quality. A rather more innovative step, however, is the parallel development of a Requirements Management System (RMS). Although only at the conceptual stage at present, such an RMS based on relational databases shows promise of not only being a valuable tool for ensuring the transparency of decision-making but also for communicating the inter-relationships of different requirements to inexperienced technical staff. The RMS will also help set priorities for NUMO’s R&D plan, which is being drafted at present and envisaged to be a continually evolving “living document”.

Many of the key R&D topics can be addressed by Japanese research organisations or as part of international collaborations. In some special areas, however, NUMO has identified special initiatives needed to ensure that essential knowledge, tools and data are available at key programme stages. Examples include:

- Establishment of an international tectonics task force to develop consensus on issues associated with volcanism and active faulting, which are particularly important in Japan
- Initiation of work to develop the next generation of PA codes & databases with the ability to better distinguish between the performance of different sites / repository designs
- Development of methodology to evaluate practical aspects of repository designs associated with constraints set by material flows and operational safety.

At present the generic MAA studies focus very much on technical aspects associated with safety and engineering practicality but, as future runs become more site-specific, environmental, socio-political and financial attributes will receive stronger weighting. Ideally, key stakeholders (e.g. members of local communities) will be able to participate directly in these exercises as a way of facilitating dialogue.

4. Conclusions

Challenging boundary conditions on the Japanese HLW programme have led to development of a very structured approach to tailoring repository concepts to the sites which may be volunteered. Iterative refinement of designs is closely coupled to assessment of the constraints set by the site and the requirements for engineering practicality and operational / post-closure safety. This structured approach is complemented by QMS and RMS, providing input to an R&D plan which aims to make best use of resources available in Japan and through collaboration with international partners.

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Derivation of unconditional clearance levels for short-lived radionuclides - a Korean approach

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Abstract. Unconditional clearance levels were derived for fifteen short-lived radionuclides. Due to the uncertainty of long-term radiological impact analysis, alpha emitting nuclides and nuclides with half-lives longer than 30 years were excluded from the scope of this study. The candidate waste streams are solid waste and waste oil generated from nuclear power reactors. The clearance levels were derived by generic assessment for enveloping scenarios, along with specific assessment for each detailed scenario such as landfill, incineration and recycling. The derived values lie in the range from 1E-2 to 1E+2 Bq/g.

1. Introduction

Clearance is a process to remove radioactive materials within authorized regulatory control from further regulatory control. The concept of clearance has been partly implemented in Korean Atomic Energy Act and relevant regulations in terms of "Self-Disposition" along with the dose criteria of 0.01 mSv/y and 1 person·Sv/y. Although Notice No. 2001-30 of the Minister of Science and Technology provides a single clearance level for thirty short-lived beta/gamma emitting nuclides as 100 Bq/g, clearance levels for other major nuclides (e.g., Cs-137, Co-60, etc.) have not been determined yet. Accordingly, a case-by-case safety review has been done for each application of clearance of waste containing nuclides not listed explicitly in the Notice. This approach requires large amount of regulatory resources and applicants' works, therefore the need of deriving unconditional clearance levels for major radionuclides generated from large-scale nuclear installations has been raised by industries. The ultimate objective of this study is to develop unconditional clearance levels for the waste contaminated with radionuclides which have not been covered by present domestic regulations.

2. Assumptions and methodology

2.1. Basic assumptions

According to the Notice No. 2001-30, the maximum radiation dose posed by the cleared materials to any individuals shall not exceed 0.01 mSv/y. In order to derive clearance levels based on the above dose criterion, a series of scenario-specific assessment should be performed. Otherwise, more generic approach, in which a set of comprehensive simple scenarios enveloping detailed ones are modelled, can be taken into consideration. In this study, both methodologies of “Specific Assessment” and “Generic Assessment” were adopted. As for the specific scenarios, landfill, incineration and recycling options were considered.

The total amount of waste cleared per year was determined to be 1,000 ton. The waste streams to be cleared were limited to solid waste and waste oil, and the target nuclides were selected as follows: H-3, C-14, Mn-54, Fe-55, Co-58, Fe-59, Co-60, Ni-63, Zn-65, Sr-89, Sr-90, Sb-125, Cs-134, Cs-137, and Ce-144. Above conditions envelop most of domestic regulatory experiences on clearance for the last 10 years. In addition, it was also conservatively assumed that the waste would not be further diluted with clean materials after clearance.

2.2. Generic assessment

In the generic assessment, a series of representative scenarios enveloping a variety of detailed ones was conservatively assumed. As a result, inhalation, ingestion, and direct exposure pathways were chosen. The inhalation scenario simulates the situation in which workers inhale contaminated dust
originated from cleared waste both in workplace and environment. In the ingestion scenario, it was assumed that workers might inadvertently swallow radioactive materials by hand-to-mouth pathway in dusty workplace (secondary ingestion) and pica children might ingest soil-like materials cleared from regulatory control. In addition, direct exposure to truck drivers and landfill workers handling cleared materials were taken into consideration.

The generic assessment was performed by spread sheet embodied macros of numerical models similar to the European Commission models [1].

2.3. **Landfill scenario assessment**

Landfill is one of the most widely used options to release materials from regulatory control. The following pathways were chosen: (1) direct exposure from buried materials, (2) inhalation of re-suspended airborne dust, and (3) ingestion of contaminated water and food.

The critical groups are (1) workers at landfill, and (2) on-site residents after closure of the burial site. Numerical calculations were performed by RESRAD [2]. In order to calculate effective dose, however, dose conversion factors (DCFs) for intake (ingestion and inhalation) were adjusted to reflect IAEA Safety Series No. 115. Taking into account of potential radiological impacts to all age groups, the standard DCFs for intake were set to the DCF values for adult multiplied by a factor of “2”. Other specific parameters for assessment are listed in Ref. [3].

2.4. **Incineration scenario assessment**

Incineration of combustible solid waste and waste oil has been partly done as an option for clearance in Korea. But some issues on re-concentration of radioactivity in the by-product (e.g., ash) have been brought up in the process of regulatory assessment. In this regard, a series of constraints were established prior to the assessment of the incineration scenario as follows: (1) Off-site incineration is allowed for waste oil contaminated only with H-3 and/or C-14; and (2) On-site incineration can be used for all waste streams, provided the area is specified as “controlled area” and ash is controlled.

It is assumed that all radionuclides existing in particulate form are still remained in the ash, but only 10% of initial radioactivity of H-3 and C-14 is remained in the ash after incineration. On the other hand, the fractions of radioactivity released to the atmosphere for H-3, C-14, and other particulate nuclides were conservatively set to 100%, 95%, and 10%, respectively. The radiological impacts due to the atmospheric dispersion of radioactive plume from the incinerator stack were calculated by INDAC within the area of radius 10 km [4].

2.5. **Recycling scenario assessment**

In this recycling scenario, it is assumed that the steel scrap can be directly reused or melted in industrial smelter or refiner and then fabricated into consumer/public products. The radionuclides in the steel scrap may redistribute into metal product, slag, and dust filter. Accordingly, the phenomena of mass partitioning and elemental partitioning in the melting process were mathematically modelled. Critical groups in the whole recycling scenario are (1) workers for collecting and transporting scraps, (2) workers at melting and fabrication plants, (3) workers for handling and transporting products, and (4) consumers using recycled products. The above scenario was numerically calculated by RESRAD-RECYCLE [5]. However, relevant DCFs for intake were also adjusted according to the same approach as the landfill scenario assessment.

3. **Calculation results and derivation of clearance levels**

3.1. **Clearance levels for each scenario**

The expected dose per unit radioactivity concentration of each radionuclide (i.e. mSv/y per Bq/g) was calculated for each scenario as discussed in Section 2. The above values were divided by 0.01 mSv/y of dose criterion and then the clearance levels for each scenario were obtained as listed in Table I. The clearance levels in Table I have been derived from the unrounded values by assigning the nearest power of 10, according to the European Commission’s approach [1].
Table I: Radioactivity concentration equivalent to 0.01mSv/y for each scenario and clearance level

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Minimum Value</th>
<th>Critical Scenario</th>
<th>Clearance Level</th>
</tr>
</thead>
<tbody>
<tr>
<td>H-3</td>
<td>7.06E+02</td>
<td>1.32E+02</td>
<td>Recycling</td>
</tr>
<tr>
<td>C-14</td>
<td>3.32E+01</td>
<td>3.10E+00</td>
<td>Incineration</td>
</tr>
<tr>
<td>Mn-54</td>
<td>1.90E-01</td>
<td>4.13E-01</td>
<td>Generic</td>
</tr>
<tr>
<td>Fe-55</td>
<td>1.58E+02</td>
<td>6.67E+02</td>
<td>Landfill</td>
</tr>
<tr>
<td>Co-58</td>
<td>2.48E-01</td>
<td>1.59E-01</td>
<td>Incineration</td>
</tr>
<tr>
<td>Fe-59</td>
<td>1.92E-01</td>
<td>9.43E-01</td>
<td>Recycling</td>
</tr>
<tr>
<td>Co-60</td>
<td>3.64E-02</td>
<td>7.60E-02</td>
<td>Generic</td>
</tr>
<tr>
<td>Ni-63</td>
<td>1.46E+02</td>
<td>7.19E+00</td>
<td>Incineration</td>
</tr>
<tr>
<td>Zn-65</td>
<td>2.45E-01</td>
<td>8.78E-01</td>
<td>Incineration</td>
</tr>
<tr>
<td>Sr-89</td>
<td>2.05E+01</td>
<td>2.03E+03</td>
<td>Incineration</td>
</tr>
<tr>
<td>Sr-90</td>
<td>1.14E+00</td>
<td>1.16E-02</td>
<td>Landfill</td>
</tr>
<tr>
<td>Sb-125</td>
<td>2.96E-01</td>
<td>1.43E+00</td>
<td>Generic</td>
</tr>
<tr>
<td>Cs-134</td>
<td>7.97E-02</td>
<td>2.00E-01</td>
<td>Landfill</td>
</tr>
<tr>
<td>Cs-137</td>
<td>1.89E-01</td>
<td>1.06E-01</td>
<td>Landfill</td>
</tr>
<tr>
<td>Ce-144</td>
<td>2.14E+00</td>
<td>7.91E+00</td>
<td>Incineration</td>
</tr>
</tbody>
</table>

It turns out that the recycling scenario is critical to H-3 and Fe-55, and the incineration scenario is dominant for C-14, Ni-63, Sr-89 and Ce-144. In addition, the clearance levels for Co-58, Sr-90 and Cs-137 were determined by landfill scenario and those for Mn-54, Fe-59, Co-60, Zn-65, Sb-125 and Cs-134 by generic assessment.

3.2. Comparison with reference values

FIG. I shows the clearance levels derived in this study, and clearance levels and exemptions levels listed in three different documents published by the IAEA [6-8]. The clearance levels proposed in this study are quite comparable to the values listed in the IAEA Safety Guide and generally lower than other reference values. The relative conservatism of the final results of this study can be mainly
attributed to the additional consideration of incineration scenario and other conservative basic assumptions such as excluding dilution effect.

4. Concluding remarks

Clearance levels for short-lived fifteen radionuclides have been derived by both generic and specific scenario assessments, based on past regulatory experiences on clearance. Korea is now planning to revise its clearance-related regulations, and therefore recently initiated a feasibility study for adopting basic principles along with values of activity concentration in the IAEA Safety Guide. In the rule-making process, the assumptions, scenarios, methodologies and results of this study may provide a clue to improve regulatory effectiveness on unconditional clearance.

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REFERENCES

Abstract. WACID, the Waste Comprehensive Information Database System, was developed to integrate the information of radioactive waste management by KINS (Korea Institute of Nuclear Safety). The system is being used to report the waste inventory and the status by the individual waste generators. The system collects the information from domestic nuclear installations through the internet into the concerted DB system. The DB system has been employed conceptual system model, unified data format and modular functional configuration. The WACID system will play an important role to direct the radioactive waste policy by Government, to promote R&D activities, to upgrade the domestic level of safe management of radioactive waste. And the WACID also will be of great role on communicating with the General Public and obtaining public understanding on any safety issues arising from every stage of waste management including final disposal.

1. Introduction

As the generation and accumulation of radioactive waste continues to increase with the domestic use of nuclear energy, there is brought up the necessity of establishing a national level comprehensive database system to which is applied the-states-of-the-art information technology in order to manage every information related to the safety management of various and massive radioactive waste sources in the systematic manner.

By the urgent national need, WACID (http://wacid.kins.re.kr) was developed to integrate the information of radioactive waste management by KINS. The system is used to report the waste inventory and the status by the individual waste generators. The system collects the information from domestic nuclear installations through the internet into the concerted DB system.

2. System architecture and design

2.1. System architecture

WACID’s Hardware is composed of main DB server, web server and application server. The main DB server and web server are separated to enhance system’s efficiency. The main DB server manages all collected input data and has a self-backup ability. We selected Oracle software to manage the DB system. Fig. 1 illustrates System structure.

FIG. 1: WACID system structure
2.2. **System module**

The DB system into 8 modules has been divided considering radioactive waste’s characteristics to facilitate incorporation of additional modules when is needed in the future. The DB system has been employed modular functional configuration for maximizing data sharing, minimizing data redundancy, enhancing the operational effectiveness and avoiding unexpected troubles due to the involvement of numerous waste generators and a variety of data characteristics from sources. The 8 modules are listed in Table 1.

<table>
<thead>
<tr>
<th>Sub module name</th>
<th>Contents</th>
<th>Sub module name</th>
<th>Contents</th>
</tr>
</thead>
<tbody>
<tr>
<td>LILW</td>
<td>Low and Intermediate Level Waste</td>
<td>LEFF</td>
<td>Liquid Effluent</td>
</tr>
<tr>
<td>RIRW</td>
<td>Radio-Isotope Radioactive Waste</td>
<td>GEFF</td>
<td>Gaseous Effluent</td>
</tr>
<tr>
<td>DDRW</td>
<td>Decommission and Decontamination Radioactive Waste</td>
<td>DISP</td>
<td>Disposition of Radioactive Waste</td>
</tr>
<tr>
<td>SNFM</td>
<td>Spent Nuclear Fuel Module</td>
<td>MISC</td>
<td>Miscellaneous Data</td>
</tr>
</tbody>
</table>

3. **Main functions**

3.1. **Data entry**

Written procedures control the data entry and verification. The data are entered electronically into the DB by various waste generators quarterly as a format of excel file. We verify the data by checking physical reliability (data formats, communication errors) and logical reliability check (limits, constraints, comparing with historical data variations). In input stage, we assure the consistency with pre-defined data format and verify the data integrity/reliability periodically. Fig. 2 shows data flow in WACID.

3.2. **Internet based system operation**

The internet based system is designed to offer the status of the nuclear waste management to the public through the web page. Also multi-user can access to the DB at the same time. The contents consist of information searching, pictures, radioactive waste dictionary, reference, community, report etc. There are three levels of access to the web site which are public user, data entry personnel and administrator. The public users can obtain only fully verified information. The data entry personnel can add and modify the individual radioactive waste information data. The administrator manages the total system. Fig. 3 shows the WACID main home page.

**FIG. 2: Data flow chart in WACID**  **FIG. 3: WACID main home page**
3.3 **Trend analysis and reports**

One of the goals of developing WACID system is to support the routine reporting of current status and future trends on radioactive waste management. To achieve this goal, WACID system has various tools for trend analysis including time series analysis. The visual reports of the analysis data are available in various formats in the system. There are various report formats such as summarized, standard, user-defined, quarterly and yearly reports. Fig. 4 shows an example of a standard report.

![FIG. 4: An example of a standard report](image)

4. **Conclusion**

We developed WACID for the purpose of not only supporting the routine reporting of current status and future trends on radioactive waste management but also integrating enormous information till now, from domestic nuclear installations through the internet into the concerted DB system. The WACID system was tested to confirm the integrity and it is now operating normally. The WACID system will play an important role to direct the radioactive waste policy by Government, to promote R&D activities and to upgrade the domestic level of safe management of radioactive waste. The WACID also will be of great role on communicating with the general public and obtaining public understanding on any safety issues arising from every stage of waste management including final disposal. In addition, the developed system will do much for achieving 5 principles (i.e., independence, openness, clarity, efficiency, reliability) of nuclear safety regulation, by providing essential information to the general public. Further extension of WACID to final disposal of radioactive waste will also be anticipated for the successive step to meet States’ responsibility, ensuring of preserving the sorted information on disposal of radioactive waste for next generation.

**Acknowledgement.** The WACID system was developed with the financial support of the Ministry of Science and Technology.
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Performance demonstration of a near surface repository for LLW and MLW in Lithuania

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Abstract. About 100 000 m\textsuperscript{3} of solid conditioned Low and Intermediate Level Wastes, generated during operation and decommissioning of the Ignalina NPP, are to be disposed in a near surface repository - a "hill"-type repository with reinforced concrete vaults and with engineered and natural barriers restricting access of water to the radioactive waste and release of radionuclides from the repository. Preliminary dose estimates confirms insignificance of releases of radioactive substances during natural degradation of the repository. The estimated doses are far below internationally accepted limit and they demonstrate high performance of the selected design and a selected site.

1. Introduction

In near future big amount of radioactive waste will arise from decommissioning activity of Ignalina NPP in Lithuania. The Ignalina NPP consists of two units commissioned in December 1983 and August 1987. Both units are graphite-moderated light-water cooled channel type Soviet RBMK-1500 reactors. Upon comprehensive assessment of technical, economic and political factors, the Unit 1 was closed in 2004 while Unit 2 will be shut down in 2009. Taking into account possible social, economical and environmental threats the Government of Lithuania approved immediate dismantling strategy, implying that about 100 000 m\textsuperscript{3} of solid conditioned Low and Intermediate Level Wastes, generated during operation and decommissioning of the Ignalina NPP, will be disposed of in the forthcoming near surface repository. A generic repository conceptual design to be applied in Lithuania was developed after scrutinizing the design and operational experience of existing near surface repositories in other countries [1] [2]. The main goal of the study is preliminary demonstration of performance of the repository built in a selected location.

2. Repository features and site-specific barriers

The basic principles of the conceptual design of the planned near surface repository are:

- a "hill"-type repository with reinforced concrete vaults should be constructed above the groundwater level because such a repository will be saturated and percolated much less and much later than a repository located below the groundwater level;
- engineered and natural barriers which should restrict the access of water to the radioactive waste and block or retard the release of radionuclides from the repository;
- a bottom bed layer beneath a cell-type reinforced concrete construction should consist of low-permeable and very low-compressible smectite clay or a mixture of properly graded silty sand, gravel and smectite clay;
- during the closure of repository the sides and top of a reinforced concrete vaults are also covered by the smectite clay barrier, the whole system being covered by a long lasting cover with an erosion-resistant top.

Various radioactive wastes are generated in the nuclear power plant and will be disposed in the repository. However, the most important waste according nuclide activities and amount are spent ion exchange resins. Before disposal the spent resins will be solidified. In a cementation plant the resins, after mixing with cement, water and additives, will be transferred into 200 L drums. The drums with waste matrix will be transferred to the cubic cells of repository. It is planned to have the form of vaults...
consisting of a number of rows of cubical concrete cells with inner dimensions of around 6 m. The wall, bottom and roof thickness is estimated at 0.5 m. If the drums will be sacked in 6 layers, each cell will contain 630 waste drums. As each vault consists of 9 cells, the total number of drums in vaults of one disposal unit will be 5670. Gaps between will be filled with grout.

Low-permeable clay barriers will be used to isolate repository from all sides and should provide a low-compressible bottom bed of the vaults with a hydraulic conductivity of less than \(10^{-10}\) m/s, and top and side seals with the same required conductivity. The clay materials must be effectively compactable so that required hydraulic conductivity is reached. Such material exists in Lithuania and has been investigated [3]. The Lithuanian Triassic clay turned out to be sufficiently rich in smectites and was proposed as main candidate for sealing of the repository. Triassic clay layers, which are more than 100 m thick, are proposed for hosting a deeply located repository in Lithuania for long-lived radioactive waste and spent nuclear fuel.

Radionuclide sorption properties in Triassic clay tested in laboratory [3] vary significantly depending on the chemical composition of the porewater solution. Alkaline plumes formed by degradation of concrete constructions of the repository could significantly impair caesium retention in surrounding soils.

3. Site characteristic and assessment methodology

Site selection process showed that no specific area in Lithuania especially favourable for radioactive waste disposal. Therefore Ignalina NPP region was considered as the region best suitable first of all as very well characterized during previous investigations. Also, its proximity to the NPP significantly reduces risk during waste transportation. The detail investigations of potentially suitable sites were concentrated in the closest vicinities of Ignalina NPP, in a distance of ~30 km from the NPP. The site selected for further investigations is a prolonged hill of Quaternary marginal till (loam) deposits. In its environs the Quaternary aquifer system of one unconfined and several confined inter-till aquifers takes place. Thickness of unsaturated zone varies from several centimeters to several meters. In local hills shallow unconfined groundwater occurs in the depth of 2-10 meters. Unsaturated zone and unconfined groundwater aquifer is lithologically composed from clayey loam and sandy loam, sand, clay, peat and silt. The proposed site is located in distance of 4 km from the NPP on a large ridge with sloping ground surface and good water run-off conditions. The site is represented by an elongated, 10-15 m high, flat-topped hill. More than 50 m thick till sandy loam with sandy inter-layers is underlying. The bottom of the hill is marshy in some places. Tributaries of Lake Drukiai – main water body draining the site - and the lake itself predetermine fast surface run-off and a good dilution conditions. Social importance of the site is very low due to small number of residents and negative demographic trends.

Groundwater significance for the performance of near surface repository in Lithuania was considered by modelling of radionuclide transport within groundwater, transfer to nearby located bog and downstream transport into Lake Druksiai. In many papers corresponding to the current understanding of migration processes of solutes in aquifers the following major migration mechanisms important for radionuclide transfer are examined: advection in unsaturated and saturated zones, molecular diffusion, hydrodynamic dispersion and retardation. The migration of radionuclides from a repository toward unconfined aquifer and eventually toward the biosphere can only start when the repository has been percolated by water infiltration and radionuclides have been dissolved by water and transported away from the waste through the engineered barriers surrounding the waste packages. All migrations mechanisms were represented using computer code AMBER [4], methodology provided in [1] and extent set of parameters both generic [5] [6] [7] and site-specific. Site-specific parameters were estimated in several field and laboratory investigation campaigns.

4. Assessment results

Potential release of radionuclides to water pathway after closure of the repository and consequent contaminants plume transport has been assessed taking into account site-specific geological and hydrogeological conditions and basing on to normal evolution scenario. This scenario describes expected evolution of the disposal system including natural degradation of engineered barriers in long-term perspective: minimal water intrusion time through the repository 150 years, advection flow through
repository for 150-250 years is 0.1×Inf. (Inf. = natural infiltration rate of 0.015 m/y), for 250-650 years – 0.5×Inf., for >650 years – 1×Inf.

Basing on simulation results the maximum activity concentration of radionuclides in groundwater at distance of 150 meters is expected: less than 11 Bq/m$^3$ of I-129 after $5×10^3$ years, less than 2000 Bq/m$^3$ of C-14 after $10^4$ years, less than 90 Bq/m$^3$ of Nb-94 after $8×10^4$ years, about $10^4$ Bq/m$^3$ of Ni-59 after $2×10^5$ years (Fig. 1). Activity concentrations of other radionuclides in groundwater are considerably lower.

**FIG. 1:** Simulated activity concentration of main radionuclides in groundwater in modelled observation well located in 150 m from repository versus time.

**FIG. 2:** Effective doses (total from all radionuclides and partial from main radionuclides) versus time for well model and for lake model (sum of external and inhalation doses and exposure from all exposure pathways).
For human exposure assessment, two simple biosphere models with typical data for the selected site have been used:

1) well model, assuming that transferred from repository to unsaturated zone (1 m thick) radionuclides are transported in the unconfined aquifer to the well of drinking water installed 150 m downstream (3 compartments of 50 m length) from the disposal system;

2) lake model, assuming that contaminant plumes in groundwater from all disposal units are discharged to the bog located 158-758 m from repository, after radionuclides are transferred to lake bay with water turnover time of 2.9 years and later left the system under consideration; local inhabitants are using lake water for fishing, irrigation etc.

In case of the normal evolution scenario the annual effective dose estimated for all radionuclides transferred to the environment do not exceed: $1.4 \times 10^{-6}$ Sv/y for the well model with a main dose forming radionuclides - I-129 and C-14 after $6 \times 10^3$ – $10^4$ years, U-234 daughters and Ni-59 after $2 \times 10^5$–$5 \times 10^5$ years; $1.2 \times 10^{-5}$ Sv/y for lake model with the same main dose forming radionuclides (Fig. 2). The estimated annual effective dose is lower than the dose constraint of 0.2 mSv for both compartments and biosphere cases.

5. Conclusion

Preliminary investigation of the most critical pathways shows that near surface repository for LLW and MLW is feasible in Lithuania and that it will meet internationally accepted safety criteria. In case of the normal evolution scenario the maximum conservative estimate of the annual effective dose is $1.4 \times 10^{-6}$ Sv/y for the well model and $1.2 \times 10^{-5}$ Sv/y for the lake model. Nevertheless, by getting new parameter estimates and by introducing new more site characterization models, the safety case should be re-assessed approaching from conservative to more realistic scenarios.

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REFERENCES


Highlights of review of L/ILW disposal in European countries, North America and Japan - institutional framework, disposal concepts, assessment approaches

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bMitsubishi Materials Corporation, Saitama, Japan

Abstract. Key information concerning L/ILW disposal in several European countries as well as in North America, has been compiled for comparison with the corresponding situation in Japan. Based on the information compiled, several observations are offered concerning the various institutional frameworks, disposal concepts and assessment approaches.

1. Introduction

Monitor Scientific has been reviewing the situation regarding LLW disposal in several European countries (Belgium, France, Finland, Germany, Spain, Sweden, Switzerland, United Kingdom) as well as in North America (Canada, U.S.A.), for comparison with the corresponding situation in Japan. In the course of this review, carried out over a number of years, a wealth of facts have been compiled concerning the institutional and regulatory frameworks as well as the performance assessment (PA) or safety assessment work carried out in support of a particular waste disposal concept.

Table 1 contains selected information accumulated during the course of the review. Further details will be provided in the poster presentation of this paper.

2. Summary observations

Based on the information compiled, several observations can be made concerning L/ILW, a few of which are noted here.

- In terms of institutional framework, the implementing agency responsible for short-lived L/ILW disposal is normally a commercial organization and independent of the government. The apparent exception to this is in Germany where the agency primarily responsible for waste management is BfS (see Table 1). However, in all cases, the regulator is independent of the implementer. Waste management funding is based on the “polluter pays” principle but enacted in different ways, e.g., in Sweden, a tax on energy consumption, or in the U.K., a tariff associated with the waste itself.

- Two main disposal concepts exist, both conforming to the multi-barrier principle:
  - (i) surface or near-surface engineered facilities - typically immobilized waste grouted in containers that are cemented in concrete monoliths, and emplaced in concrete modules; and
  - (ii) mined rock caverns. In both cases, cement/concrete is widely used within each repository.

- Institutional controls are incorporated in surface / near-surface disposal site waste management; such controls consist of a combination of active and passive surveillance.
<table>
<thead>
<tr>
<th>COUNTRY</th>
<th>Implementer; Regulator; Key Regulations</th>
<th>Relevant Waste Classification</th>
<th>Waste Disposal Concept(s)</th>
<th>Performance Assessment (PA) Scenarios / Exposure Pathways</th>
<th>Institutional Control Period / Comment / Key Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>BELGIUM</td>
<td>Agency for Radioactive Waste and Enriched Materials (ONDRAF/NIRAS); Federal Agency for Nuclear Control (FANC), Vinçotte Nuclear Association (AVN); Law of 15/4/1994, Royal Decree of 20/7/01.</td>
<td>Category A: low-level, short-lived; low- and medium-activity waste, β-γ radionuclides with $t_{1/2} \leq 30$ years; trace amounts of α-β-γ radionuclides with $t_{1/2} &gt; 30$ years.</td>
<td>Multi-barrier (MB); near-surface concrete vaults. No LLW repository yet; CILVA central storage facility (Mol-Dessel); 3 local partnerships established for possible disposal site selection;</td>
<td>Generic PA involving groundwater and atmospheric (gaseous releases) pathways.</td>
<td>Safety Analysis Report structure based on USA system and NRC document NUREG-1199 requirements (see USA below) [1][2].</td>
</tr>
<tr>
<td>CANADA</td>
<td>Low-level Radioactive Waste Management Office of AECL (LLRWMO - federal); Ontario Power Generation (OPG - Ontario); Nuclear Safety Commission (CNSC); Nuclear Safety and Control Act (2000); CNSC Regulations (P-290 on policy).</td>
<td>LLRW: Defined by exclusion − neither spent nuclear fuel nor mill/mine tailings; further classified into Historic waste (LLRWMW - federal) and On-going waste (nuclear waste producers/owners, e.g. OPG).</td>
<td>MB; No national or provincial LLW repository yet; Memo of Understanding signed with local community for long-term management, deep-rock cavern (~200-400 m) preferred option in Ontario for LLW.</td>
<td>Reference Scenarios (expected evolution), base case - groundwater pathway only; Intrusion Scenario via drilling of boreholes – direct exposure only.</td>
<td>No formal period but assessment assumed human intrusion after 300 years, based on potential (deep) disposal concept. Assessment period extends beyond 10,000 years [3].</td>
</tr>
<tr>
<td>FINLAND</td>
<td>Teollisuuden Voima Oy (TVO); Radiation and Nuclear Safety Organization (STUK); Nuclear Energy Act (990/1987), Decision on General Regulations (398/1991), YVL Guides.</td>
<td>LLW: Low-level (typically ≤ 1 MBq/kg); Medium-level (typically 1 MBq/kg to ≤ 10 GBq/kg).</td>
<td>MB; Near-surface VLJ repository (1992-), site of Olkiluoto nuclear power plant (NPP); Lovisa (1998-) at NPP; both near-surface (70-100 m) rock caverns with reinforced concrete silo(s).</td>
<td>FL; three types of scenario; basic (conservative data); realistic (less pessimistic data), disturbed evolution (loss of one or more barriers). Groundwater pathway to biosphere.</td>
<td>No institutional period considered necessary for this depth [4].</td>
</tr>
<tr>
<td>FRANCE</td>
<td>National Agency for Radioactive Waste Disposal (ANDRA); Nuclear Safety Authority (AVN); Règles Fondamentales Sûreté I.2, II.2e.</td>
<td>Type A: mainly radionuclides with $t_{1/2} &gt; 30$ years; small amounts of radionuclides; maximum α-activity &lt;370 MBq/t (in repository); &lt;3.7 GBq/t per package, as of end of institutional period.</td>
<td>MB; Centre de la Manche, 1969-1994, formerly trenches, now near-surface concrete vaults; Centre de l’Aube, 1992-; near-surface concrete vaults.</td>
<td>Normal scenarios (barriers perform as expected), groundwater pathway only – well through release plume; Accident scenarios (one or more barriers fail), inadvertent human intrusion – construction / agriculture; Atmospheric pathway for gaseous C-14/H-3 to nearby home.</td>
<td>Assessment of scenarios covers operational phase, institutional control period (overall 300 years) and long-term confinement [5].</td>
</tr>
<tr>
<td>GERMANY</td>
<td>Federal Office for Radiation Protection (BfS) via Organization for the Construction and Operation of Repositories for Waste (DBE); Federal Ministry of Environment, Nature Protection and Nuclear Safety (BMU – federal), Länder (States); Atomic Energy Act (1959), Federal Mining Law (1980) Safety Criteria (1983).</td>
<td>Konrad (site-specific): Non heat-generating waste; activity limits derived separately for each waste package; wastes divided into five types: solids, metals, cementitious, concentrates, bituminized.</td>
<td>MB; Asse (~ 800 m), 1967-1978, former salt mine, now closed; Morsleben (~ 500 m), 1994-1998, former salt mine, now closed; Konrad (~ 1000 m), former iron ore mine, under review by court following challenge.</td>
<td>Safety assessment for normal operations and assumed incidents during operational and post-operational phases; long-term safety evaluated for release of radionuclides into water pathways; Irrigation pathway (farming) is key exposure pathway.</td>
<td>No control measures considered necessary – “maintenance free”. Assessment modelling out to 10⁶ years [6].</td>
</tr>
</tbody>
</table>
### Table 1 (continued): Key highlights from review of LLW disposal in European countries, North America and Japan

<table>
<thead>
<tr>
<th>COUNTRY</th>
<th>Implementer; Regulator; Key Regulations</th>
<th>Relevant Waste Classification</th>
<th>Waste Disposal Concept(s)</th>
<th>Performance Assessment Scenarios / Exposure Pathways</th>
<th>Institutional Control Period / Comment / References</th>
</tr>
</thead>
<tbody>
<tr>
<td>JAPAN</td>
<td>Nuclear Fuel Limited (JNFL); Atomic Energy Commission (AEC); Atomic Safety Commission (NSC); Atomic Energy Basic Law, Waste Management Law (2000).</td>
<td>LLW; also LLRW containing comparatively high activity (reactor core components e.g. control rods)?</td>
<td>Multi-barrier (MB); Rokkasho, near-surface concrete vaults with covers of bentonite/sand mixture layer and surface soil layer.</td>
<td>Migration Scenario (sand permeability for cement barriers, groundwater control by covers) – Ingestion of fish (likely), Dwelling with surface well (less likely), Land-use Scenario (construction / residence) – Cover soil source (likely).</td>
<td>~ 300 years [7]. [7]</td>
</tr>
<tr>
<td>SPAIN</td>
<td>National Radioactive Waste Management Company (ENRESA); Council on Nuclear Safety (CSN); Nuclear Energy Law (1964); Law 15 (1980), CSN Guides GS-9.1, GS-9.2.</td>
<td>Low/Intermediate level $\beta$$\gamma$ radionuclides $t_{1/2} \leq 30$ years; long-lived $\alpha$-activity $\leq 3.7$ GBq/t.</td>
<td>MB; El Cabril (1992-); near-surface concrete vaults</td>
<td>Groundwater pathway – dwelling with well through release plume; Inadvertent human intrusion.</td>
<td>300 years [8].</td>
</tr>
<tr>
<td>SWEDEN</td>
<td>Nuclear Fuel and Waste Management Company (SKB); Nuclear Power Inspectorate (SKI) / Radiation Protection Institute (SSI); Act on Nuclear Activities (1984), SKI-FS 1998:1, SSI-FS 1998:1.</td>
<td>Short-lived ($t_{1/2} \leq 30$ years) LLW and ILW (operational waste).</td>
<td>MB; SFR-1 - near-surface rock cavern ($\sim 60$ m), with reinforced concrete silo for ILW</td>
<td>SAFE: Base Scenario (same climate); also Barrier defects, Climate change, Human activities. Primarily groundwater pathway (well); gas phase transport (C14) considered low consequence relative to groundwater pathway.</td>
<td>No institutional control period considered necessary at this depth; assessment period at least 10,000 years [9].</td>
</tr>
<tr>
<td>SWITZERLAND</td>
<td>National Cooperative for the Disposal of Radioactive Waste (NAGRA); Federal Nuclear Safety Inspectorate (HSK); Nuclear Energy Act (2003), HSK Guidelines (R-14, R-21).</td>
<td>Short-lived ($t_{1/2} \leq 30$ years) low- and intermediate-level LLW: $&lt; 20$ kBq/g; primarily operational waste.</td>
<td>MB; mined rock cavern, deep disposal (500-700 m); No LLW repository yet; local Canton voted against Wellenberg site; site feasibility being re-investigated.</td>
<td>Reference Scenario (expected behavior) - primarily groundwater pathway; gas-induced release also evaluated; Alternative Scenarios (erosion, human activities, gas phase release).</td>
<td>No institutional control period; safety analysis was technically acceptable by authorities [10].</td>
</tr>
<tr>
<td>UNITED KINGDOM</td>
<td>British Nuclear Fuels plc (BNFL - Drigg), Atomic Energy Authority (AEA - Dounreay); Environment Agency (EA) / Scottish Environmental Protection Agency (SEPA); Radioactive Substances Act 1993, EA+SEPA Guidance Document G1A.</td>
<td>LLW: &lt; 4 GBq/tonne $\alpha$, &lt; 12 GBq/tonne $\beta$$\gamma$.</td>
<td>MB: Drigg, 1959-, near-surface, formerly trenches, now concretelined structures (1995-).</td>
<td>Central Projection Scenario (continuous evolution - regional glaciation, coastal erosion, valley glaciation); Future Human Actions and Disruptive Events (discrete events). Groundwater and gaseous pathways to biosphere.</td>
<td>100 years, assuming closure at 2050. Conditional risk evaluated, assessment timeframes: 0-250 y; 250-10,000 y; &gt; 10,000 y [11].</td>
</tr>
</tbody>
</table>
The disposal concepts discussed here are for the most part relevant to short-lived L/ILW. However, waste classification differs between countries and does not necessarily conform explicitly to IAEA recommendations based on radioactivity and half-life [12]. In particular, the North American method of classifying LLW is by exception. Irrespective of waste classification, waste acceptance criteria are linked directly to a site-specific assessment. Some (the more recent) assessments address both the radioactive and toxic nature of the waste; other assessments address only the radioactive component.

Performance Assessment / Safety Assessment methodologies carried out over the past 10 years or so follow similar approaches, conforming for the most part to IAEA’s methodology described in [13].

Most recent site selection activities, e.g., in Belgium, Canada, are being carried out in close cooperation with local communities. This is considered the optimum approach for engendering local community support and progressing successfully with waste disposal projects.

REFERENCES


The national plan for radioactive waste and recoverable material management in France

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Abstract. The development of a National Plan for Radioactive Waste and Recoverable Material Management is a priority for the French Nuclear Safety Authority. The main objective is to search management solution for each type of radioactive waste produced. Radioactive waste without management solution should be clearly identified and management solutions should be proposed in a reasonable time schedule. The plan should take into account principles applying for waste in general and also radiation protection principles. The Nuclear Safety Authority oversees the production of the plan which is submitted periodically to a plenary group where stakeholders involved in radioactive waste management are represented. Draft recommendations are discussed during the meetings of the plenary group since the end of 2004. A draft of the National Plan for Radioactive Waste and Recoverable Material Management is available since July 2005 and the plan, in its final version, may constitute an appendix of the law on high level radioactive waste which should be discussed by the Parliament in 2006. The trends of the plan may also be approved by an article of the law.

1. Introduction

The need to develop a comprehensive plan to manage radioactive waste was identified a few years ago. One of the objectives of such a plan is to improve transparency, effectiveness and efficiency in the field of radioactive waste management.

2. Context

Further to a request from the Parliamentary Office for the Assessment of Scientific and Technological Options (OPECST), on the basis of the report produced in 2000 by the deputy of the Drôme department, Mrs. Michèle Rivasi, the Nuclear Safety Authority (ASN) confirms that it is in favour of drawing up a national plan for radioactive waste management. During a presentation to the Council of Ministers on 4 June 2003, the Minister for Ecology and Sustainable Development stated her intention to produce such a plan. On behalf of the public authorities, the ASN was tasked with overseeing its production.

This proposal is in conformity with a provision already included in article L.541-11 of the Environment Code (resulting from law 75-633 of 15 July 1975 concerning the disposal of waste and recovery of materials). This article gives the Minister for the Environment the option of drawing up national disposal plans for waste considered to be particularly harmful or requiring special treatment and storage. This option was, for example, used for waste contaminated by polychlorinated biphenyls (PCB).

For radioactive waste, a more global framework appeared necessary, to allow consistent management of all radioactive waste, guaranteeing safe management and the corresponding financing, in particular for its disposal, by determining the relevant priorities.

The following were invited to take part in the work on the National Plan for Radioactive Waste and Recoverable Material Management: representatives of the waste producers, the disposal facilities, the National Agency for Radioactive Waste Management (ANDRA), environmental protection associations, elected representatives and the directorates of the ministries concerned.

3. Goals of the national plan for radioactive waste and recoverable material management

The Plan is based on the knowledge of different types of radioactive waste described in the reference document “National Inventory of Radioactive Waste and Recoverable Material” which was published...
in November 2004 by the National Agency for Radioactive Waste Management (www.andra.fr). This inventory enables the quantities of waste produced to be estimated for various time-frames, including 2010-2020. The goals of the National Plan for Radioactive Waste and Recoverable Material Management are presented below:

- clear definition of the waste to be considered as radioactive, taking account of the existence of natural radioactivity of variable levels and of certain radioactive materials for which reuse has not been envisaged;
- search for management solutions for each category of radioactive waste produced;
- taking charge of older radioactive waste which has been "forgotten";
- consideration of the concerns of the public, who rightly or wrongly are worried about the fate of radioactive waste;
- the consistency of the entire radioactive waste management structure, whatever the level of radioactivity or the chemical or infectious toxicity, in particular for waste with a "mixed" risk;
- optimisation of waste management by the waste producers: nuclear industry, more conventional industries (in particular those using naturally radioactive substances but for their other properties), activities using radionuclide sources, medical sector, earth taken from old polluted sites, mining industry (uranium mines in particular);
- consistency of practices to deal with polluted sites and reclamation methods;

leading to clear, meticulous and safe management.

The National Plan for Radioactive Waste and Recoverable Material Management does not aim to duplicate the inventory work done by the National Agency for Radioactive Waste Management. It will therefore be more particularly based on the information already available in this framework. It is not however impossible that this plan could bring to light certain waste that does not appear in the inventory, in particular through a more detailed definition of radioactive waste.

4. Principles of the national plan for radioactive waste and recoverable material management

The development of a National Plan for Radioactive Waste and Recoverable Material Management should take into account principles regarding waste management and also radiation protection principles, as:

- principles of justification, optimisation and limitation required by the radiation protection regulations;
- mitigation of the production of the waste and limitation of the toxicity of the waste;
- responsibility of the producer until the waste is safely treated and disposed;
- information of the public;
- identification of radioactive waste management routes and prevention of inadvertent effect on the Environment and on human health;
- mitigation of transport;
- development of solutions for radioactive waste without liable producer available.

5. Interface with research into high-level long-lived waste

For high-level long-lived waste, research into disposal channels is governed by law (article L.542 of the Environment Code, resulting from the law of 30 December 1991), which requires that a report on the progress of research into the disposal of high-level long-lived waste be presented to Parliament before the end of 2006, so that a debate can be held on the follow-up to be given to this research, which has intensified and diversified since the 1991 law.

Producing a National Plan for Radioactive Waste and Recoverable Material Management does not interfere with this process, which solely concerns high-level long-lived waste. The National Plan for Radioactive Waste and Recoverable Material Management above all meets the need to provide channels for managing and disposing of waste which does not fall into this category, such as disused sealed sources, waste containing radium, graphite waste, dismantling waste, and so on. However, producing it at the same time as the Government's report requested in article L.542 of the Environment
Code will give the political decision-making bodies an overview of radioactive waste problems and will place the special case of high-level long-lived waste in a more general context.

6. Initial conclusions

A first version of the national plan for radioactive waste was established and presented to the members of the plenary group in September 2004. Other versions were examined by the plenary group during the beginning of 2005. Long term management solutions or research programs to establish disposal routes exist for the main part of the radioactive waste. For some waste, like disused sealed sources, investigation should be conducted to determine long term management solutions. The application of the principle of justification that came into force in the Public Health Code could lead the Government to ask the removal of a great number of radioactive sources (smoke detectors, lightning conductor) which should be properly managed. Disposal routes for radioactive waste produced by the decommissioning program of the first generation of nuclear power plants should be determined, especially for graphite waste. The mission of the National Agency for the Management of Radioactive Waste consisting in recovering and storing radioactive waste from private individuals or establishments without the resources to dispose of it should be recognised as a mission of public utilities. It would also seem important to monitor the consistency of the regulatory provisions concerning radioactive waste and the benefits of requiring a declaration from all radioactive waste producers need to be examined.

7. Prospects

The initiative consisting in producing the National Plan for Radioactive Waste and Recoverable Material Management was on the whole warmly received by the various parties involved, including the representatives of activities which are not among those the public authorities normally find themselves faced with in this field. It should be noted that internationally, this approach was seen as a good practice, in particular within the framework of the meeting to review the national reports drafted under the terms of the joint convention on the safety of spent fuel management and the safety of radioactive waste management, which took place in Vienna on 3 to 14 November 2003. Production of a National Plan for Radioactive Waste Management in each country was recommended in the final report issued by the review meeting.

A new version of the Plan was available in July 2005 and it can be downloaded from the Nuclear Safety Authority website (www.asn.gouv.fr). In its report in March 2005 on radioactive waste management, the Parliamentary Office for the Assessment of the Scientific and Technological Options (OPECST) announced that the Plan should constitute an appendix of the law on the management of high level radioactive waste which could be debated by the Parliament in 2006. The trends of the National Plan for Radioactive Waste and Recoverable Material Management national plan for radioactive waste may be approved by an article of the law.

The ASN considers that developing the National Plan for Radioactive Waste and Recoverable Material Management is a priority and that it will eventually lead to more open, more exhaustive and safer management of radioactive waste in France.

Appendix 1: Classification of radioactive waste in France

<table>
<thead>
<tr>
<th>Activity</th>
<th>Period</th>
<th>Very short-lived (half life &lt;100 days)</th>
<th>Short lived (half life &lt;30 years)</th>
<th>Long-lived (half life &gt;30 years)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Very low level waste</td>
<td>Management by radioactive decay</td>
<td>Dedicated surface repository</td>
<td>Recycling channels</td>
<td>Surface disposal (Aube repository) except disused sealed sources (under investigation)</td>
</tr>
</tbody>
</table>
### Appendix 2: Inventory of the solutions for the management of radioactive waste

<table>
<thead>
<tr>
<th>Level of Hazard</th>
<th>Producer/owner of the waste</th>
<th>Exist</th>
<th>Does not exist but research are conducted (and/or) channel under investigation</th>
<th>Does not exist, no research or channel under investigation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Low (VLLW)</td>
<td>Conscious, solvent</td>
<td>VLLW from operating or dismantling of nuclear installations, included some waste contaminated by tritium</td>
<td>Technologically enhanced normally occurring radioactive waste (solvent industry)</td>
<td>Does not exist, no research or channel under investigation</td>
</tr>
<tr>
<td></td>
<td></td>
<td>One part of mining residues</td>
<td>Waste and effluent from research sector</td>
<td>smoke detectors</td>
</tr>
<tr>
<td></td>
<td>Not conscious, but solvent</td>
<td>Technologically enhanced normally occurring radioactive waste (not identified yet)</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Conscious or not, not solvent</td>
<td>Waste from polluted sites contaminated by radioactive material, one part of waste from mining of uranium ores (…). Technologically enhanced normally occurring radioactive waste from past activities</td>
<td></td>
<td></td>
</tr>
<tr>
<td>High (&gt;LLW)</td>
<td>Conscious, solvent</td>
<td>Waste from operating nuclear installations short-lived, ILW and LLW (and waste produced by sectors outside the Nuclear energy industry) Some disused sealed sources Short-lived radioactive waste from medical and research activities Waste with multiple hazards (radioactive, chemical or biological)</td>
<td>long-lived LLW from nuclear industry (graphite) One part of radium bearing waste Long-lived ILW and HLW from nuclear industry (research are conducted in compliance with article L.542 of the Code for the Environment) Majority of waste contaminated by tritium Disused sealed sources Liquid organic ILW</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Not conscious, but solvent</td>
<td>Possibly old sources not inventoried, (sets for education …)</td>
<td>old sources not inventoried</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Conscious or not, not solvent</td>
<td>Some waste produced by the clean up of polluted sites</td>
<td>One part of radium bearing waste</td>
<td>One part of radium needles, radium items, lightning conductors containing radium</td>
</tr>
</tbody>
</table>
Probabilistic evaluation of uncertainties of long-term impacts through the computer code GSRW-PSA

H. Kimura, S. Takeda, M. Munakata

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Abstract. In order to assess the long-term safety of geologic disposal, it is essential to evaluate various uncertainties of long-term radiological consequences associated with geologic disposal of radioactive wastes. JAERI has developed the probabilistic safety assessment code for HLW disposal system (GSRW-PSA) to estimate the uncertainties in parameters and conceptual models. The computer code system GSRW-PSA is based on a groundwater migration scenario, and consists of a set of sub-modules for sampling of model parameters, calculating the release of radionuclides from engineered barriers, calculating the transport of radionuclides through the geosphere, calculating radiation exposures of the public, and calculating the statistical values relating the uncertainties and sensitivities. The results uncertainty analyses show that the conceptual model uncertainties of geologic media (homogeneous or heterogeneous) and also the uncertainties of solubility limits to the maximum flux are large.

1. Introduction

The Japan Atomic Energy Research Institute (JAERI) has developed the probabilistic safety assessment code system GSRW-PSA (Generic Safety assessment code for geologic disposal of Radioactive Waste – Probabilistic Safety Assessment) [1], to estimate the uncertainties in parameters and conceptual models. The computer code system GSRW-PSA is based on a groundwater migration scenario, and consists of a set of sub-modules for sampling of model parameters, calculating the release of radionuclides from engineered barriers, calculating the transport of radionuclides through the geosphere, calculating radiation exposures of the public, and calculating the statistical values relating the uncertainties and sensitivities. This paper summarizes the results of uncertainty analyses for geologic disposal of high-level radioactive waste (HLW) using the computer code system GSRW-PSA.

2. Uncertainty analyses

The GSRW-PSA intend to evaluate the potential radiological consequences of geologic disposal of HLW is based on a generic approach rather than site specific one, reflecting from the current situation in Japan. However, it might be required to define the disposal system to some extent so as to enable the analysis of transport of radionuclides in geologic formations and the biosphere following entry of groundwater into a repository and release of them to surrounding strata. A crystalline bedrock is now considered to be one of potential geologic strata for geologic disposal of HLW in Japan. We thus assume that a repository will be constructed in a deep and stable granite bedrock at a depth of 1,000 m. The rock is described in terms of two major hydraulic units; fractured zones and rock mass. It might be reasonable to assume that the repository is constructed in a stable rock mass having enough distances from fractured zones as shown Fig.1, in order to avoid the occurrence of a short circuit of groundwater from the repository to the biosphere. In this paper, we assume that the granite rock mass surrounding the repository has relatively low hydraulic conductivity and the fractured zone has high hydraulic conductivity like weathered granite as shown Fig.2. The migration pathways from the repository to the surface were estimated by Monte Carlo simulations of groundwater flow analysis (2D finite element analyses) assuming that the hydraulic conductivities of the rock mass and fractured zone are based on the PDF (Probability Density Function) obtained from Fig.2. Two type of Monte Carlo simulations were carried out, one is that the granite rock mass and the fractured zone are homogeneous hydraulic properties, and another is that those are heterogeneous ones. Estimated migration pathways of
homogeneity case are shown in Fig.3. Also Monte Carlo simulations of radionuclides transport were carried out base on the obtained migration pathways.

**FIG.1:** Location of a potential HLW repository and fractured zone

**FIG.2:** The distribution of measured hydraulic conductivities of granite in Japan

**FIG.3:** Estimated migration pathways from the repository to the surface
Solubility limits of elements are key parameters to control release rates of radionuclides from the repository. Variability of solubility limits are estimated by Monte Carlo simulations of geochemical analysis (EQ3/6[2] code) using major geochemical database (EQ3/6-TDB[3], JNC-TDB[4]) and assuming the variability of chemical conditions of groundwater. Figure 4 shows the CCDFs (Complementary Cumulative Distribution Function) of Se solubility limits based on the various geochemical conditions.

**FIG. 4:** The CCDF of Se solubility limits based on the various geochemical conditions.

3. **Results and discussion**

The radionuclide transports through the granite rock mass and fractured zone are calculated by 1D LTG (inverse Laplace Transform Galerkin method) model in the GSRW-PSA taking account of the effects of matrix diffusion assuming the empirical low $2b = 2\sqrt{T}$ ($b$: fracture width, $T$: transmissivity of rock). Figure 5 shows the CCDFs of maximum flux of Se-79 from the engineered barrier, and figure 6 shows the CCDFs of maximum flux of Se-79 from the granite rock mass. The maximum flux of Se-79 from the engineered barrier have ranges of more than 6 orders of magnitude, and it means that the uncertainties of solubility limits to the maximum flux are large. The maximum flux of Se-79 from the granite rock mass have ranges of more than 9 orders of magnitude, and it means that the conceptual model uncertainties of geologic media (homogeneous or heterogeneous) and also the uncertainties of solubility limits to the maximum flux are large. These features are confirmed by the PRCC (Partial Rank Correlation Coefficient) profile of Se-79 as shown in Fig.7.

**FIG. 5:** The CCDF of maximum flux of Se-79 from the engineered barrier.
FIG. 6: The CCDF of maximum flux of Se-79 from the granite rock mass

FIG. 7: The PRCC of maximum flux of Se-79 from the granite rock mass.

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REFERENCES

Legislation of the clearance system

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Abstract. This paper summarizes the legislation of a clearance system for radioactive waste in Japan. The Nuclear and Industrial Safety Agency (NISA), a regulatory body in Japan, studied the legislation of the clearance system in the Radioactive Waste Safety Subcommittee (RWSS). The RWSS determined to employ the IAEA’s safety standard, “Application of the Concepts of Exclusion, Exemption and Clearance (RS-G-1.7) [1],” as the Japanese clearance levels and prepared a final report on the roles and responsibilities of the government and the operators with respect to clearance and on the technical requirements for clearance verification. NISA submitted a bill for fiscal 2005 to the Diet for amending the “Law for the Regulations of Nuclear Source Materials, Nuclear Fuel Material and Reactors (LRNR)” in accordance with the report. NISA will prepare rules for clearance levels and their verification methods and put them in place by the end of 2005.

1. Introduction

The Japan Atomic Power Company’s Tokai Nuclear Power Plant, the first commercial nuclear power plant in Japan, permanently ceased its operation and removed spent fuels. It started decommissioning in December 2001. As the decommissioning of the power plant progresses, “wastes which are not radioactive” or “wastes which are not necessary to be treated as radioactive materials” are likely to be generated. Therefore, it is imperative to establish a system for discriminating wastes which are not necessary to be treated as radioactive materials from radioactive wastes by the clearance levels.

Under these circumstances, the RWSS of the Nuclear and Industrial Safety Subcommittee studied a highly reliable and adequately operable verification system for clearance levels, and fundamental matters of technical requirements for the verification system in order to legislate the clearance system. The RWSS prepared its final report taking into consideration the basic ideas of clearance levels and the verification methods of clearance levels that had been introduced by the Nuclear Safety Commission (NSC).

NISA submitted a bill for fiscal 2005 to the Diet for amending the LRNR for legislating the clearance levels and verification methods based on the RWSS report.

2. Clearance levels

The NSC prepared a report titled “Clearance levels in main nuclear reactor facilities” (hereinafter called “Clearance level report”) in November 2000, and derived scientifically the standard values of clearance levels for solid materials (concrete and metals) in nuclear reactor facilities (light-water reactors and gas reactors) in Japan. Furthermore, the NSC reviewed the clearance levels taking into consideration RS-G-1.7 and new findings after the “Clearance level report,” and prepared a new report titled “Concentrations of radioactivity in wastes that are not necessary to be treated as radioactive materials out of wastes produced from the dismantling of nuclear reactor facilities and nuclear material utilization facilities.

Considering the review results by the NSC, and from the viewpoints of international consistency, potential international mobility of cleared wastes, versatile applicability of regulations to wastes other than those from the dismantling of reactors, and easy understanding of regulations, it is anticipated that, in setting up the clearance levels, the values shown in RS-G-1.7 will principally be the regulating values of the government for wastes produced from the decommissioning of nuclear reactor facilities.
3. Verification system for clearance levels

The “verification of clearance levels” is a procedure that, firstly, the nuclear operators identify a waste as the “waste that is not necessary to be treated as a radioactive material” using the clearance levels, and then the government takes additional considerations to the judgement of the operators.

3.1. Flow of clearance level verification

In the decommissioning of reactor facilities, works are carried out step by step from dismantling of equipment and buildings to shipping the dismantled pieces. The probable verification practices in the decommissioning process are shown in Fig. 1. The same practices will also apply to solid waste materials (metals and concrete) produced during the operation of nuclear reactor facilities.

3.2. Basic idea on the roles of the government and the operators in the clearance level verification

A waste object that is verified to be below the clearance level is not necessary to be treated as a “nuclear fuel material or/and waste contaminated by nuclear fuel material” stipulated in the LRNR. Therefore, it is important that the reliability of verification is ensured by the proper confirmatory considerations taken by the government in addition to the judgement by the operators. For that purpose, the government requires to stipulate technical standards for measurements and judgement of waste objects. The government also requires confirming the adequacy of the methods for measurement and judgement and the results of measurements and judgement implemented in advance by the operators in accordance with the technical standards.

On the other hand, the operators must properly dispose of wastes produced during operation under its own responsibility, provide methods for measurements and judgement of waste objects, properly sort out waste objects in accordance with its own methods, and measure the concentration of a radioactive nuclide. Thus, the operators determine if the waste is below the clearance level. Furthermore, the operators must properly secure the waste determined to be below the clearance level, keeping it away from foreign objects or contamination. The operators must record the results of the measurements, etc. and keep those records properly.

3.3. Verification by the government

Involvement by the government in verification is roughly divided into two steps: the first is that the government confirms (approves) the adequacy of “methods for measurements and judgement of waste objects” submitted by the operators; and the second is that the government verifies “the waste object to meet the clearance level” that was determined in accordance with the methods approved by the government.

The operators must prepare the “methods for measurements and judgement of waste objects” in accordance with the technical standards stipulated by the government. As these methods are important for ensuring the quality of measurements and judgement, the government will approve the adequacy of the methods prior to actual measurements. The descriptions in the documents for measurements and judgement of waste objects will possibly include the following: method for selecting radioactive nuclides to be assessed, setting up their composition ratios, setting up of measurement conditions and method for measurements depending on the characteristics of each waste object, method for evaluating results of measurements, method for recording the waste object whose measurements and judgement have been completed, method for keeping the records, and preparation of quality assurance program.

“Results of measurements and judgement of waste objects” obtained by the operators are those obtained by the methods for measurements and judgement approved by the government. In principle the government will carry out verification based on the records. However, from a viewpoint of enhancing the objectiveness and reliability of verification, the government may carry out sampling inspection if necessary. Furthermore, the government timely checks whether the quality assurance activity for clearance level verification is properly implemented by the operators. Verification of “results of measurements and judgement of waste objects” will be partly carried out by JNES that was established for supporting the nuclear safety regulation activity.
For a waste object that is determined to be below the clearance level in the verification carried out by the government, it is considered that the government will issue a certificate that proves the waste meets the clearance level.

4. **Voluntary efforts by operators**

The operators expressed their commitment that they, well aware of their responsibility and obligation as the waste-producing operator, will ship cleared wastes to waste treatment contractors who recognize that the wastes come out of the nuclear power facilities or to limited industrial waste disposal sites until the clearance system is rooted in society. Also, they will take the lead in promoting the recycling of waste while obtaining public understanding. These efforts by the operators will be effective for smooth fixation of the clearance system. Specifically, making the first consignees identifiable in the disposal or recycling process of the wastes, for example, disposal sites in the case of burial disposal and interim disposal businesses when the wastes are recycled as valuables, until the clearance system is rooted in society. Establishing the framework of grasping and recording the whereabouts of wastes will lead to the enhancement of the public confidence in the verification system.

5. **Conclusion**

Objects of clearance are wastes that are produced from activities inevitable to the people’s living, and the significance of clearance, premised on the verification of safety, will pave the way for waste
disposal including appropriate and rational recycling, effective utilization of national resources, and reducing loads on the environment. This idea coincides with that of Japan which aims at promoting formation of a recycling-oriented society in the 21st century.

In recognition of this, the systematization of clearance levels and verification procedures is under way by NISA in co-operation with JNES, and substantial technical standards will be prepared as ministerial ordinances on the basis of the idea shown herein. Public comments were solicited on the RWSS’s report, and input from the participants in symposiums on the clearance system was obtained. Several of these comments and input were reflected in the final report. We will work for legislating the clearance system by incorporating a wide range of public opinions.

REFERENCE

Progress in Japan's TRU waste disposal technologies in the generic research and development phase

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Abstract. In accordance with the stepwise approach to disposal of TRU waste in Japan, the utilities and JNC have been collaborating to promote associated generic R&D based on the technical achievements in the HLW disposal program, and to prepare a second progress report with support from related organizations such as RWMC, JNFL and CRIEPI. In order to build confidence in the technical feasibility of the disposal system, R&D activities have focused on the understanding and development of realistic near-field models and databases, taking into account the specific design components and phenomena encountered in a TRU waste disposal system. The main aim of the second progress report is to demonstrate more rigorously and transparently the technical feasibility and reliability of the specified disposal concept and to provide input for the future implementation and regulatory framework of TRU waste disposal prior to the start of operation of the first commercial reprocessing plant in Japan.

1. Introduction

In Japan, large volumes of low-level radioactive waste (LLW) containing nuclides with long half-lives, such as transuranic (TRU) nuclides, fission products (e.g. iodine-129) and activation products (e.g. carbon-14), will be generated by the operation and decommissioning of domestic and overseas reprocessing plants and a domestic MOX fuel fabrication plant. Such wastes are termed "TRU waste". The activity levels of this waste range over a wide spectrum from intermediate to very low, depending on the waste type, and it is immobilized in various materials including cement, organic substances (e.g. asphalt), metal, etc.

Promoting the safe disposal of all waste types, including TRU waste, is recognized as being one of the key requirements for utilizing the nuclear fuel cycle. As generators of TRU waste, the Federation of Electric Power Companies (FEPC) and the Japan Nuclear Cycle Development Institute (JNC) have responsibilities this task.

One of the features of the associated R&D program is that its progress has been documented at appropriate intervals, with a view to clearly demonstrating the level of achievement and identifying further R&D issues. As a major milestone, FEPC and JNC collaborated to produce a first progress report on the disposal concept for TRU waste in Japan ("1st TRU report"[1]) in March 2000, with the assistance of RWMC (Radioactive Waste Management Funding and Research Center), JNFL (Japan Nuclear Fuel Limited), CRIEPI (Central Research Institute of Electric Power Industry) and JAERI (Japan Atomic Energy Research Institute). The report summarizes the research results to date on safe TRU disposal concepts in Japan. The Atomic Energy Commission (AEC) established the TRU Sub-Committee under the Advisory Committee on Nuclear Fuel Cycle Backend Policy, with the task of evaluating the feasibility of the TRU waste disposal in Japan and discussing open R&D issues taking into consideration the technical achievements documented in the 1st TRU report.
Against the background of the findings of the TRU Sub-Committee, FEPC and JNC have continued to pursue R&D aimed at enhancing the reliability of the safety assessment of the disposal concept and establishing a more rational approach to disposal.

Prior to the start of operation of the first commercial Japanese reprocessing plant, where test operation by JNFL started in December last year, FEPC and JNC decided to publish the second progress report (2nd TRU report) with the aim of providing timely technical input for both future implementation and for the regulatory framework for TRU waste disposal. This paper summarizes the results of the 2nd TRU report.

2. Estimation of properties and generated volumes of TRU waste up to around 2050

The volume of TRU waste generated from the operation and decommissioning of domestic reprocessing and MOX fabrication plants, considering a full operation period, and returned wastes from the COGEMA and BNFL reprocessing plants up to around 2050 is estimated to be about 140,276 m³, assuming application of treatment methods currently planned. TRU waste is categorized into 3 groups based on nuclide concentrations and properties namely waste destined for geological, mid-depth and shallow land disposal. For waste intended for mid-depth and shallow land disposal, the criteria set out in the regulations for the LLW from nuclear power plants were taken into consideration. However, the nuclide composition of the LLW originating from NPPs is different to that of TRU waste, and revised criteria for specific nuclide concentrations for TRU waste were therefore formulated using the same calculation methods as for the LLW from the NPPs. The results show that about 60% of TRU would be suitable for shallow land disposal and about 20% would be destined for mid-depth disposal.

TRU wastes for geological disposal were grouped into 4 types. Group 1 includes weak-sorbing nuclides, such as I-129. Group 2 includes hulls & end-pieces, which generate heat and contain large concentrations of C-14. Group 3 includes chemical substances such as sodium nitrate, which have to be analysed carefully in terms of impact on radionuclide migration. Group 4 consists of other miscellaneous wastes (Table 4).

<table>
<thead>
<tr>
<th>Disposal concept</th>
<th>Volume, m³</th>
<th>Group</th>
<th>Volume, m³</th>
</tr>
</thead>
<tbody>
<tr>
<td>Geological disposal, depth greater than 300 m</td>
<td>26,641</td>
<td>Group 1</td>
<td>318</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Group 2</td>
<td>6,732</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Group 3</td>
<td>6,175</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Group 4</td>
<td>13,416</td>
</tr>
<tr>
<td>Mid-depth disposal, depth 50-100 m</td>
<td>25,205</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Shallow land disposal, depth several meters</td>
<td>88,431</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>140,276</td>
</tr>
</tbody>
</table>

3. Engineered barrier system (EBS) design and layout of the repository

The EBS designs for LLW from the NPPs can be applied for the TRU categories “Shallow land disposal” and “Mid-depth disposal”. For the EBS for geological disposal of groups 1 and 2, a combination of a compacted bentonite barrier and a cementitious backfill were considered with the purpose of retarding the migration of I-129 and C-14. The content of I-129 and C-14 in groups 3 and 4 is relatively small and thus only cementitious material was selected as the main EBS component. In addition, group 3 includes large amounts of sodium nitrate, which might reduce the distribution
coefficients for iodine and carbon on cement paste [2]. To avoid such an influence, the cavern for group 3 was separated sufficiently from the caverns for the other groups, particularly groups 1 and 2. Based on the reference concept and layout, detail designs in several rock properties (crystalline and sedimentary) and detail assessments of system evolutions including long-term stability, heat and temperature etc were performed. Schemes on operation and closure of the repository facility were also examined taken into account applicable techniques and technologies.

A concept of "co-located" repository HLW and TRU waste within a single complex is also studied (Fig. 1). It is expected to reduce the cost of the repository and burden of siting effort for geological disposal. There is concern that water with a high pH and/or high nitrate content leaching from the TRU wastes could adversely change the properties of barriers in the HLW repository.

Based on modellings studies with conservative assumptions, it was concluded that adverse effects could be avoided by separating the two facilities by a distance of a few tens to a few hundred metres.

![Image](image.png)

**FIG. 1: Birds-eye view of an example concept for co-location with HLW**

4. Safety assessment of TRU waste disposal

Firstly, important FEPs that could potentially affect nuclide migration in the repository environment were considered, taking into account Japanese geological environments and NEA’s international FEP list as follows:

- chemical composition of groundwater;
- longevity of engineered barrier materials, particularly the interaction of cement/bentonite;
- high-pH plume effect on host rock;
- hydraulic conditions in the near-field;
- colloid effects;
- effects of natural and artificial organic materials such as asphalt, cellulose, etc.;
- microbial effects;
- radiation effects;
- effect of sodium nitrate;
- impact of gas generation.

Based on the results of evaluating these FEPs, nuclide migration analyses were performed using the same geological model as in the H12 report [3] with more realistic near-field models and databases taking into account the specific design components and phenomena encountered in a TRU waste disposal system in the reference case. A top-down sensitivity analysis for uncertainties remaining in the deterministic analysis was adopted. I-129 in group 1 gave the dominant dose, but the value was much smaller than the regulatory limits set in foreign countries. The hydraulic properties of the natural barrier were the key parameters for the groundwater scenario. The results confirmed that higher containment waste forms for group 1 could result in a more robust repository concept, taking into account a range of geological conditions.
A comprehensive safety assessment, based on a "Safety Case", was conducted in accordance with international practice [4,5].

5. Conclusions

The concept of disposal of TRU waste in Japan appears to be technically feasible and the research and development effort described in the 2\textsuperscript{nd} TRU report shows its reliability and confidence than 1\textsuperscript{st} TRU report. Whilst there were still uncertainties and a number of outstanding issues where further work was required, there were no insurmountable obstacles to making safety cases for the disposal of TRU waste in Japan. Emphasis of R&D programme should continue to be placed on increasing safety and confidence in geological disposal so that the outcome can contribute to the incremental decision making process in a transparent and equitable manner.

REFERENCES


Considerations for near surface disposal for wastes of non-power activities in Malaysia

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Abstract. Malaysia needs to have a near surface disposal facility to dispose wastes produced from non-power activities. A site must be chosen to locate such a facility and must be suitable to accept the type of wastes produced in Malaysia. MINT can conduct a well plan study which consist of different expertise starting from the initial design of the disposal site until the final detailed design stages. The design is much dependent on the site specific characteristics. The impact of such facilities on the human environment has to be kept as low as reasonably achievable and the safety assessments of the facilities must satisfy the Atomic Energy Licensing Board of Malaysia (AELB) requirements and guidelines and be in general compliance with the various international standards or guidelines such as of the IAEA. From scientific point of view, the technical merits of the site should have high importance in the site acceptability decision. The better the merits of the site, the greater the probability that it can be shown to satisfy the performance objectives and the safety requirements. If the merits of the site do not follow a scientific point of view, future stability of the site to extend over hundred of years will be in doubt. The public and the government should not compromise with any decision if safety would be neglected.

1. Introduction
Disposal of radioactive wastes is to isolate these wastes from the human environment until the radiation hazard has decreased to safe level. For most solid wastes and solidified liquid wastes, the long-term isolation and containment provide the primary protection. In Malaysia, sources of radioactive wastes comes from different activities. Being a developing nation, the main sources which contributed a high volume of wastes are the technically enhanced natural radioactive material (TENORM) wastes from the oil mining and minerals activities. There are also wastes from radioisotopes application, non-power activities such as industry, medical, research and development, etc., which actually contributed a smaller volume of wastes. All these are benefit activities which contributed to the economic growth of Malaysia. Radioactive wastes which consist of different characteristics such as consisting of radioisotopes with different half life, radioactivity, physical and chemical properties, etc, must be kept away from man for the duration of their potentially harmful levels of radioactivity. A near surface disposal for the type of wastes especially from non-power activities is likely to offer such solutions. However, for the TENORM wastes which are of high volume, usually the relevant authority such as the AELB and DOE recommended the producers to dispose off the wastes on their expenses with proper approval on the design and disposal management of the wastes into disposal site. MINT can conduct a well plan study which consist of different expertise starting from the initial design of the disposal site until the final detailed design stages. The design is much dependent on the site specific characteristics. Then the post operational activities must be conducted such as environmental monitoring for a certain period of institutional control. Surveillance of the site is also needed for a predetermined number of years. The impact of such facilities on the human environment has to be kept as low as reasonably achievable and the safety assessments of the facilities must satisfy the Atomic Energy Licensing Board of Malaysia (AELB) requirements and guidelines and be in general compliance with the various international standards or guidelines such as those of the IAEA.

2. Storage against disposal
As Malaysia is a country with low volumes of wastes and with limited resources for disposal and facility development, storage can offer an acceptable interim solution to the waste management. The waste can be retrieved from storage and eventually sent to final disposal. However, with rapid development, accumulation of wastes from 25 years operation at MINT and prediction of higher
volumes of wastes produced in future, a final solution must be found for the proper management of 
wastes. This is also to overcome the double handling of waste (storage and eventual disposal) 
accompanied by higher overall costs in future and exposures to workers and the public. There are 
additional criteria that must be fulfilled in addition to the standard criteria for a disposal site. Detailed 
design of the disposal sites with the intention to permanently dispose off the wastes and after a certain 
period of institutional control these sites are expected to be returned to their natural environment. The 
criteria for disposal sites are more stringent than for storage sites.

3. Site selection/characterization

Site chosen to locate a facility plays a dominant role in determining the acceptability of that waste 
storage or disposal facility. The current waste disposal practices for Low Level Wastes involve near 
surface shallow ground burial. In Malaysia, which is situated in the Equatorial region with high 
rainfall and undulating landscape with high groundwater level, a suitable site must be selected. The 
natural characteristics of the site and the engineered safety design features of the facility should 
complement each other. The degree to which the reliance can be placed on the geosphere to retard the 
radionuclide migration from the facility depends on the disposal concept chosen, the nature of the 
radioactive waste and the characteristic of the site. Extensive screening and characterisation on the site 
can give a best possible site selected. However, when site selection is constrained to a specific area or 
region or, for a site with limited radionuclide retardation capabilities, greater reliance must be placed 
on the engineered barriers designed into the facility.

In order to have a suitable site, MINT needs to conduct a site selection/characterisation programme for 
a disposal facility with the fundamental objectives to:

1. select a suitable site;
2. characterise the site to satisfy licensing/regulatory requirements and guidelines set by AELB 
of Malaysia;
3. provide relevant scientific data for safety assessments and the long term performance 
predictability;
4. provide design feedback for the basic and detail engineering design of the facility;
5. provide site monitoring as necessary.

The main ground based disposal concepts are:

1. Shallow land burial:
   1.1 minimum improvement to trench disposal, 
   1.2 highly engineered design.
2. Shallow rock cavity:
   2.1 engineered facilities in rock with depth ranging from 50 to 500 m, 
   2.2 existing rock cavities of intermediate depth such as disused mines, etc.

Shallow land burial seems to be the suitable choice for radioactive wastes from the non nuclear cycle 
activities from which MINT has accumulated wastes for a long time, and also for future wastes 
production in Malaysia.

4. Criteria and considerations

A large area may or may not be suitable based on an analytical overview of the region of interest. 
Some illustrative examples of exclusionary criteria are listed below and such areas should be excluded:

1. Malaysia cannot assume that it is free from any volcanic activities in the future. Effects from 
recent events such as the tsunami which occurred in 11 countries around the Indian Ocean, and 
frequent volcanic tremors, areas of high seismic activity and other tectonic processes such as 
faulting and folding near the western coastal region of Sumatra and Java, should be excluded 
and a suitable site free from all such activities must be chosen;
2. wetlands and areas within a 100-year floodplain;
3. areas where mass wasting, erosion, slumping or land sliding can occur with relatively high 
frequency;
4. largely populated areas or areas of anticipated urban expansion;
5. areas of unique economic potential such as mineral exploration;
6. preserves of unique flora and fauna to conserve the natural environment;
7. areas with high rainfall;
8. areas with a high potential of fire.

Then the criteria on accessibility, geology and hydrogeology can be considered. The areas left after the application of exclusionary criteria can then be put through extensive screening and characterisation studies. Political and socio-economic considerations may also influence the initial steps of the site selection process. The relative importance of such considerations can vary enormously from region to region. For nations with a high population density and small area, such factors gain importance.

The characterisation studies on the preferred site involve an extensive process consisting of field and laboratory studies, modelling studies and the hydrogeological monitoring. The site analysis and overall safety assessment then provide an answer as to the technical suitability of the site.

The considerations in the site selection/characterisation can be broadly divided into two groups: technical considerations, and political and socio-economic considerations.

From a scientific point of view, the technical merits of the site should have high importance in the site acceptability decision. The better the technical merits of the site, the greater the probability that it can be shown to satisfy the performance objectives and the safety requirements. The choice of a site should not be exclusively political. If the merits of the site do not follow a scientific point of view, the future stability of the site to extend over hundred of years is in doubt. The public and government should not compromise with any decision if safety is neglected.

5. **Host media**

The choice of the host media for disposal facilities depends on the type of terrain available, the disposal concept, the hazardous nature of the waste, the degree of isolation required and the economics of the project. For low level wastes, burial in shallow ground in surface deposits may be adequate. The host media can include clay deposits, clay/sand deposits, or sand deposits.

6. **Technical requirement for a disposal site**

The primary requirement of disposal facilities based on different host media is the same: the hazard to humans and the environment must fall within acceptable levels. Since radionuclide release scenarios from the facility to water pathways are the most probable, hydrological characterization of the site (surface water and groundwater studies) is thus of paramount importance.

A site should also satisfy the following requirements:

1. Capable of being characterised, modelled, analyzed and monitored.
2. Provide sufficient depth to water table so that groundwater intrusion into the waste will not occur. For disposal concepts that involve locating facilities below the water table, the molecular diffusion should be the only predominant means of radionuclide movement. The total disposal system must satisfy the overall criteria.
3. The surface drainage at the site from any upstream should be minimal.
4. Provide a sufficient distance to the groundwater discharge point, preferably, the residence times of radionuclides over this path length should be large.
5. Not to be located where nearby facilities or activities could effect the ability of the site to meet its performance objectives.

The technical requirement for the disposal site for licensing purposes should satisfy the local regulatory body. Other desirable features may be related to social and political acceptability of the site and the economic considerations such as the existence of access road, etc. The most important considerations of the site for disposal facilities based in shallow ground, are the hydrological and geochemical considerations and complete information on the surface hydrology, hydrogeology and geochemistry is required.
7. Safety assessment

Safety analysis of the whole system (the site, facility and the environment) provides answers to the acceptability of the disposal system including the suitability of the site. Safety assessments with mathematical models and simulated accidental scenarios are necessary because the time frame of safety assessments extends into hundreds of years. Safety assessments end with a consequence analysis for each scenario and provide an estimate of resulting dose to humans. Often the licensing requirements for disposal facilities include the stipulation that the resulting dose to humans will be below certain pre-set limits.

Assessment of data

Assessment was carried out by first identifying the radiological criteria issued by the Atomic Energy Licensing Board (AELB) and other relevant technical bodies including the International Atomic Energy Agency (IAEA) to be used as the basis and guide in the whole exercise. The criteria used were 1 mSv/y for the individual member of the public and 1 man.Sv for the collective dose of the overall population.

Critical group or groups of the population must be identified who will be affected directly or indirectly by the radiological exposure from the proposed near surface disposal site. The following step was to identify the source term through which various critical exposure pathways may occur, which may finally lead to a significant exposure received by the population. Based on exposure pathways, the most appropriate dosimetric model can be developed or available relevant commercial model used to estimate the amount of dose delivered. The estimated annual dose received by the population was then compared with the criterion 1 mSv/y for the individual member of the public in order to determine feasibility of the site to be used as a disposal site.

From the analyses, site specific guidelines can be derived for allowable residual concentration of radionuclides in soil, calculation of risks and guidelines values. It can also be used to reduce residual radioactivity to levels that are as low as reasonably achievable. Appropriate site specific parameters must be used as much as possible so as to be realistic so that will reasonably ensure that individual dose limits and or constraints will be achieved. To build confidence to the AELB and the public, sensitivity and uncertainty analysis can be conducted on every parameter, especially data of parameters which are not site specific so that the uncertainty can be verified.

Pathways for radionuclide transport from the facility to humans in general can include routes via air, soil and water. Intrusion by humans deliberately or not into repository provides another potential pathway. Water pathways are generally considered to be probable, and are thus more important for human exposure. These include surface water and the groundwater transport. The climate of the area, the topography of the site, the nature of the sediments and the hydrogeologic characteristics of the site have strong influence on these pathways.

The surface water pathway can be important during the operational phase of the facility or in the long term through the erosion of earth cap by surface runoff. The infiltrating precipitation can enter the engineered repository through cracks in the concrete cap or through the water table rise (for open bottom facilities). The leached radionuclides are transported by groundwater flow to surface water bodies from where dose to humans can occur through ingestion of fish, vegetation, animals or direct contact. The groundwater flow, characteristics and the geochemical retardation (through sorption/desorption, ion exchange) thus play an important role in attenuating the migration of radionuclides to the biosphere.

In developing the hydrogeologic model of the site, it is necessary to calibrate the model with the experimental measurements of hydrogeologic parameters. An appropriate numerical model capable of describing the physics of the groundwater flow system is an integral part of the safety analysis based on mathematical models of various components of the system. The overall safety assessments also provide some feedback as to the site related parameters in terms of where the accuracy of the site parameters can be improved.
8. Conclusion

Near surface disposal for the wastes produced in Malaysia especially from the non-power activities is likely to offer solutions. A site must be chosen to locate such a facility and must be suitable to accept the type of wastes produced in Malaysia. MINT can conduct a well planned study which consists of different expertise starting from the initial design of the disposal site to the final detailed design stages. The design is much dependent on the site specific characteristic. Then the post operational activities must be conducted such as environmental monitoring for a certain period of institutional control. The impact of such facilities on the human environment has to be kept as low as reasonably achievable and the safety assessments of the facilities must satisfy the Atomic Energy Licensing Board of Malaysia (AELB) requirements and guidelines and be in general compliance with the various international standards or guidelines such as those of the IAEA. From a scientific point of view, the technical merits of the site should have high importance in the site acceptability decision. The better the merits of the site, the greater the probability that it can be shown to satisfy the performance objectives and the safety requirements. The choice of a site should not be exclusively political. If the merits of the site do not follow the scientific point of view, future stability of the site to extend over hundreds of years will be in doubt. The public and government should not compromise with any decision if safety is neglected. Public acceptance of the public living near the disposal site must be gained to avoid delay of the project.

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Feasibility study of monitoring and its technology for geological repositories

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Abstract. This paper describes a systematic review of the study in regard to the monitoring for geological repositories of the HLW (High-Level Waste) and its technical feasibility. The results of the studies are now being formulated into the systematic diagrams which show the relations between each monitoring data and its utilization in understanding the behaviour of the geological disposal system. Also, the technical study has been conducted focusing on measuring technologies and signal transmission technologies considering deep underground conditions.

1. Introduction

Monitoring of geological disposal has been discussed in relation to the post closure phase with a period of institutional control. It is a widely accepted principle of disposal that, “The management of radioactive waste should, to the extent possible, not rely on long term institutional arrangements or actions as a necessary safety feature, although future generations may decide to utilize such arrangements, for example to monitor radioactive waste repositories or retrieve radioactive waste after closure has been effected. (IAEA, 1995)” However, even though passive safety assurance has been an essential concept, it is still desirable to prepare various active or passive institutional control methodologies. One of such active institutional control methodologies has gradually gained widespread acceptance in international studies. IAEA describes as one of objectives of monitoring “to strengthen understanding of some aspects of system behaviour used in developing the safety case for repository” (IAEA (2001). As an important way of enhancing public confidence in geological disposal, it may be the implementation of monitoring that extends from the early stages of the repository development programs to a definite period after the repository has been closed.

RWMC has been studying the roles of monitoring and monitoring technologies. The results of the studies on monitoring technologies are now being formulated into the systematic diagrams which show the relations between each monitoring parameter and its utilization in understanding behaviour of geological disposal system which can contribute to the future planning of the repository management. This systematic diagram which is called “technology menu” will be refined to contribute to monitoring programs prepared by various organizations and agencies as well to find and clarify the key technical points to be developed for preparing the reliable monitoring system of geological repository.

2. Technology menu as integration of monitoring related discussion

The technologies applicable for monitoring of geological disposal should be selected while taking into consideration the diverse monitoring objectives. Also, the different roles and objectives of implementing organization, safety regulators, local governments and other interested parties should be taken into account when monitoring plan will be in actual implementing steps. The many different objectives (needs) must be accommodated when monitoring is implemented in the future. In considering preceding discussions on the objectives of monitoring which have been discussed at international view by OECD/NEA and IAEA, the monitoring technologies in geological disposal is to be considered looking at not only technical aspects but also a variety of information including sociological aspects.

With this in mind, the feasibility study on monitoring technologies based on the survey of the existing technologies has been initiated with the discussion of the possible application of information in applying each monitoring measure to the current repository concept. The result of the studies are now
being formulated in preparing the "technology menu" that explains the structural way of how to interpret monitoring data into information utilizing for understanding repository condition by asking the so-called "5W1H" elements: Why (objective); Who; When (from/to); Where; What; and How (methodology). This approach increases the flexibility in developing the monitoring programs and selecting desirable monitoring measures at different agencies following their roles and objectives.

FIG. 1: Structuring flowchart development of the technology menu (Torata et al., 2004)

The efforts to prepare the technology menu for monitoring of geological disposal were based on the investigative methodologies shown in Figure 1. Shown in the chart of Figure 1 is the way to understand how the behaviour of the disposal system component will be analyzed and explained and be connected with the possible contributing data from applicable monitoring methods. The chart will be extended to show the technical option to be chosen for the monitoring programmes in accordance with various monitoring purposes. In discussing monitoring data and availability based on the existing technologies, the key technical subjects are indicated which should be further developed toward application of monitoring with much reliance. The current goal is to provide specific and appropriate information which will be able to be utilized in making monitoring programmes considering the technical characteristics of geological disposal, as well as knowledge obtained through an extended technology development.

3. Basic technology surveys and studies

With regard to the technologies ("seeds") that provide the basis for monitoring of geological disposal, the surveys and studies have been conducted focusing on the measuring technology (sensors) which must be developed to be applicable in considering their use under geochemical environment.

One of the latest trends in measuring technologies is a sensing combined with fiber optics, which are increasingly being used to measure factors such as temperature, pressure and deformation. But the geochemical sensor which can be applied to deep underground conditions has still not been found through technical surveys (Takegahara et al: 2004).
A study is also under way on the signal transmission technologies that will be applied to the design of the monitoring system without significant influence on the repository safety such as producing water path to be the potential radioactivity release. These studies on reliable sensors and signal transmission should be pursued in parallel with considerations on applying these technologies and equipment at the deep underground conditions.

Moreover, further studies are now prepared for improvement in such technical subjects as heatproof, pressure resistance, radiation resistance, long-term operational capability (including power supply) and longer-distance transmission. In conducting improvement and development, current requirements are fixed as specified in the Figure 2, establishing highly and reliable technology which can be applied to the deep underground conditions.

FIG. 2: Example of environmental conditions of geological repository

FIG. 3: Concept of underground wireless transmission system in geological disposal repository
In RWMC, the underground wireless transmission system has been kept developing, applying low frequency electromagnetic waves. From the perspective of suitability for underground wireless transmission, the use of such type of waves is considered the most applicable, based on actual results obtained under conditions that resemble monitoring of geological disposal, such as the data transmission at ground subsidence, and applications in civil engineering. Experiments by RWMC have been already conducted at the Hard Rock Laboratory (HRL) in Sweden under the courtesy of SKB, where it was confirmed experimentally that basic characteristic of an existing apparatus be preferable in underground rock formation. Theoretical studies are being conducted to prepare design basis supposing different characteristics of repository configurations with several transmission media.

Figure 3 presents a concept of an underground wireless transmission system in geological repositories in the future. Also, it shows the sensors installed in the drifts of geological repositories which transmit monitoring data to the ground in terms of underground wireless transmission system through relaying system (for example, development of network sensing system and software for relay).

4. Conclusions

The fundamental study of monitoring technologies is under way preparing technology menu as the comprehensive knowledge which will contribute to the monitoring programmes. The extended technology development is going to enhance reliability of monitoring technologies through experimental confirmations in Underground Research Laboratories (URLs). These activities on summarizing technical knowledge are aiming to show proper and applicable technical option supporting for the agents who plan repository monitoring and to enhance understanding of some aspects of system behaviour used in developing the safety case for repository.

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Safety report for the Central interim storage facility for radioactive waste from small producers

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Abstract. The Central Interim Storage for institutional radioactive waste in Brinje near Ljubljana, Slovenia was put into operation in 1986. In 1999 its operation was transferred to ARAO - the national Slovenian Agency for radwaste management. Simultaneously with the transfer of operation ARAO also received a request from the regulatory body to reconstruct the storage and to provide the Safety report (SR) for the facility. ARAO faced two main problems in the course of realization of the regulatory provision: firstly, the extent of the reconstruction works far exceeded the dimensions originally expected and requested by the provision and, secondly, the safety report for the facility still did not exist. Due to these facts the reconstruction was much more demanding and the preparation of the safety report more complicated than expected. In this paper the SR is presented, with emphases on the description of safety analyses for the Central Interim Storage and the consequent operational conditions and limitations.

1. Introduction

In 1999 the Agency for Radwaste Management took over the management of the Central Interim Storage (CIS) in Brinje, intended only for radioactive waste from industrial, medical and research applications in Slovenia. With the transfer of the responsibilities for the storage operation, ARAO, the new operator of the facility, received also a request from the competent regulatory body in Slovenia - the Slovenian Nuclear Safety Administration (SNSA) - for refurbishment and reconstruction of the storage and for preparation of the safety report for the storage with the operational conditions and limitations.

In order to fulfil these requirements ARAO first thoroughly reviewed the existing documentation on the facility, the facility itself and the stored inventory. It was found that some installations and systems in the storage facility were out-of-date and had fallen into disuse, the inventory was poorly characterized and arranged, the system of procedures and instructions was limited and not suitable for the extent of public service of radioactive waste management. Based on the findings of this review, ARAO prepared several basic documents for improvement of the current conditions in the storage facility.

In between the year 2000 and 2003 ARAO prepared several most important documents: plan for refurbishment and modernization of the CIS providing an integral approach, project documentation for renewal of electric installations, water supply and sewage system, ventilation system and the safety report for the reconstruction of the facility. In 2003 and 2004 all the works were done and the safety report for the commissioning period was prepared taking into account all improvements and changes introduced by the refurbishment and reconstruction of the facility. Based on relevant regulations and new atomic act, the following chapters of the SR were prepared:

1. The safety approach to the LILW storage,
2. Description and location analysis of the Central interim storage,
3. Technical characteristics of the Central interim storage,
4. Safety analysis of the Central interim storage,
5. Organizational measures for construction and pre-operational testing,
6. Organizational measures for commissioning and normal operation of the Central interim storage,
7. Operational conditions and limitations,
8. Ionizing radiation protection service, its methods and equipment,
9. Radioactive waste management and disposal,
10. Review of the plans, measures and procedures to prevent radiological accidents,
11. Quality assurance program,
12. Review of the measures for physical protection of the LILW storage and stored radioactive
   waste,
13. Planned measures and necessary equipment for closure of the Central interim storage.

2. Central interim storage in Brinje after remediation

The Central Interim Storage facility in Brinje is the only facility in Slovenia for LIL solid radioactive
waste from small producers (medicine, industry and research activities). The facility was constructed
in 1984 and put into operation in 1986. The interim storage facility is a near-surface concrete building
covered with soil. The building is subdivided by concrete walls into nine storage sections and an
entrance area. One section for radioactive waste is deepened compared to the floor of the other
sections, and is intended for more active spent sources. The ventilation system in the facility is
renovated and incorporates 3 different filters including the HEPA filter, thus enabling reduction of
radon concentration and air contamination with daughter products. A specially designed water and
sewage collecting system retains all liquids within the closed system in the sump; liquids are
discharged after the measurements of the radioactive contamination, which has to be below the
limitations. The storage facility is physically protected by an alarm system, which is connected to the
24-hour security service. Fire protection is assured by smoke detectors which are placed over each
section and connected to the fire alarm.

The ground plan of the facility remains 10.6 m x 25.70 m with a height of 3.6 m. Currently around 60
m³ of solid waste is held in the storage, giving the equivalent of 280 210-l drums. The total activity of
the waste inventory is estimated at 2400 GBq (end of 2004). The major contribution to the total
activity comes from the disused teletherapy source of Co-60 with present activity around 1800 GBq.
While the safety analysis was made for the current volume and activity of stored waste, the capacity of
the CIS in the safety report is limited to 330 drums (210-l) of conditioned waste.

3. Safety analyses for the central interim storage facility

The most important part of the safety report is safety analysis, which was performed for normal and
abnormal operational conditions by taking into account the external and internal events which could
influence storage safety. Based on the types of stored radioactive waste, packaging characteristics,
construction of the storage facility together with the location properties and anticipated normal and
abnormal operational conditions, several potential events were identified. By using elimination
criteria, sets of relevant events were defined with relevant scenarios which were then analyzed.

During the normal operational conditions in the storage two different scenarios were analyzed.
Assessment of external irradiation due to the storing of radioactive waste was done on the basis of the
regular monitoring of the storage and its surrounding. Results of the measurements demonstrate that
the dose rate due to gamma radiation from the storage decreases to the natural background within 10 to
20 m from the storage entrance.

Analyses of atmospheric gas release due to emission of radon and its daughter radioisotopes from the
storage are based on the conservative Gauss model of gas release. The results of the calculations show
that the radon concentration is approximately 3 - 5 Bq/m³ at a distance of 50 m from the storage
entrance, i.e. at the reactor centre fence, which is negligible even when compared with the natural
radiation of soil.

For the abnormal operational conditions eleven different potential scenarios were screened (Table 1).
Based on the elimination criteria, which are defined by waste characteristics, location, facility
characteristics and operation (procedures and instructions), only two different scenarios were relevant
and selected for further analyses.
Table 1. Scenarios for abnormal operational conditions for the central interim storage facility

<table>
<thead>
<tr>
<th>Potential scenarios</th>
<th>Elimination criteria</th>
<th>Relevant scenarios</th>
</tr>
</thead>
<tbody>
<tr>
<td>Spilling of the liquid in the storage</td>
<td>• Waste characteristics</td>
<td></td>
</tr>
<tr>
<td></td>
<td>• Facility (technical system)</td>
<td></td>
</tr>
<tr>
<td>Drop and crush of a waste package during the manipulation</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Direct irradiation with radioactive waste</td>
<td></td>
<td>Dissipation of the waste</td>
</tr>
<tr>
<td>Direct contact with radioactive waste</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Explosion in the storage</td>
<td>• Administrative procedure</td>
<td></td>
</tr>
<tr>
<td>Explosion in nearby facility</td>
<td>• Location (distance)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>• Facility (construction)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>• Administrative procedure for nearby facilities</td>
<td></td>
</tr>
<tr>
<td>Fire</td>
<td></td>
<td>Fire</td>
</tr>
<tr>
<td>Flooding (groundwater level rise, rainwater, ..)</td>
<td>• Location (measurements)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>• Facility (technical systems)</td>
<td></td>
</tr>
<tr>
<td>Earthquake</td>
<td>• Location (measurements)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>• Facility (construction)</td>
<td></td>
</tr>
<tr>
<td>Aeroplane crash</td>
<td>• Location (as regards to other facilities)</td>
<td></td>
</tr>
<tr>
<td>Terrorist attack</td>
<td>• Administrative procedure (physical protection)</td>
<td></td>
</tr>
</tbody>
</table>

Dissipation of the waste includes three different possible scenarios: drop and crush of a waste package during the manipulation, direct irradiation with radioactive waste and direct contact with radioactive waste. All three scenarios, which are relevant also for the storage facility, result in the same consequences: irradiation of the workers. Calculation of the doses to the workers showed that in the most serious scenario the effective dose would not exceed 1 mSv.

According to the study on fire safety in the storage facility, minor confined fire may possibly occur in the area intended for workers and in the control area where also combustible waste (or the packaging) is placed. Analyses of the fire in the typical drum in the control area - prepared by using the IAEA procedure for assessing radionuclide concentrations in air in case of fire - show that the maximum equivalent dose to the security personnel due to the fire would be 265 micro Sv. The dose contribution to the nearby living resident would be 3 micro Sv. Additionally, several administrative measures have been undertaken for fire safety assurance together with regular physical inspections every two hours.

4. Operational conditions and limitations

Based on the safety analyses of normal and abnormal operational conditions the radiological influence of the storage facility for workers, population and the environment was calculated or measured, and is given in Table 2. It could be seen that the calculated and measured values are far below legislative limitations given for workers or for the population. Based on the assessment of equivalent doses from radiological impact of the storage facility, the following operational limitations are accepted:

- the annual individual effective dose to a member of the critical group should not exceed 0.1 mSv;
- the annual individual effective dose for a worker in normal and abnormal operational conditions should not exceed 10 mSv.
Table 2: Radiological influence of the storage facility

<table>
<thead>
<tr>
<th>Group</th>
<th>Equivalent dose (mSv/year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Workers - Regular radiological monitoring</td>
<td>0.25 (individual dose for 50 hours)</td>
</tr>
<tr>
<td>Workers - Acceptance of radioactive waste</td>
<td>2.5 (individual dose at acceptance rate of 50 packages per year)</td>
</tr>
<tr>
<td>Workers – Inhalation of radon</td>
<td>0.09 (individual dose for 200 hours per year)</td>
</tr>
<tr>
<td>Administrative worker at the Reactor centre</td>
<td>0.024 (inhalation due to gas release)</td>
</tr>
<tr>
<td>Security personnel</td>
<td>0.004 (inhalation due to gas release) 0.007 (external irradiation due to the storage)</td>
</tr>
<tr>
<td>Farmer</td>
<td>0.0002 (only inhalation due to gas release)</td>
</tr>
</tbody>
</table>

Due to the storage characteristics, the load of the ground floor should be administratively controlled and must not exceed the maximum load. One section has to remain empty to allow sufficient space in case of emergency. The storage facility has to be ventilated before entering, in order to achieve equilibrium concentration of radon and daughter products.

5. Conclusions

The preparation of the safety report for the Central Interim Storage facility was very demanding from several different points of view. The main problem arose due to the fact that the storage was operating and the waste had been stored for 15 years, while the safety report still did not exist. Additionally, the format and the content of the safety report for storage facility was not prescribed, thus giving room for different interpretations and opinions. All this resulted in the fact that ARAO needed nearly 2 years to conclude the safety report and to obtain expert opinion.

Safety analysis proves that the facility together with the stored waste poses no threat to the environment. The safety report states that there are none of the normal and abnormal operational conditions expected, which would radiologically affect the staff, local population and the environment. That applies for the phase during which reconstruction works were performed and for the commissioning phase, which is foreseen to last for maximum 24 months. Based on the result of the measurements during the commissioning period, the final revision of the safety report will be prepared, which will include all the changes to the facility and also some improvement of safety analyses in accordance with the new situation to obtain the operational license.

REFERENCES


Cover model of national radioactive waste repository at Mochovce site

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Abstract. National low and intermediate level radioactive waste repository is operating from 2001 and it is appointed for disposal of solid and solidified low and intermediate level radioactive waste resulting from the operation and decommissioning of nuclear facilities NPP Jaslovské Bohunice and NPP Mochové, from decommissioning of NPP A1 in Jaslovské Bohunice, waste form research institutes, laboratories, hospitals and other institutions involved in activities during which radioactive waste are produced. The near-surface disposal site at Mochové is designed a vault-type concrete structure housing of 7200 concrete fibre reinforced containers as the final waste packages with accumulated volume of 22 320 m3. After operational period repository will be covered by structure of final covering, which must warrant repository safety at least during the institutional control period (300 year). To prove long-term stability of final repository covering the methodology with activities described in this paper was proposed.

1. Introduction
After the operational period the repository will be covered by a structure of final covering, which must warrant repository safety at least during the institutional control period (300 years). To prove long-term stability of the final repository, covering the methodology with the following activities was proposed:

1) Selection of sealing natural materials for construction of final covering and its geotechnical parameters experimental measurement;
2) Physical modelling of periodical weathering influence to radiological characteristics of soils selected for construction of final covering;
3) Construction of in-situ small-scale physical model of cover and long-term monitoring of selected parameters;
4) Mathematical modelling of radiological characteristics of repository covering.

2. Description of project tasks
2.1. Selection of natural sealing material for cover model
The natural sealing material should fulfil the following parameters:

- hydraulic conductivity \( k \leq 10^{-9} \) m/s,
- size distribution curve should fulfil requirements,
- maximal particle size should not exceed 20 mm,
- liquidity limit \( w_L \) should not exceed 50%,
- plasticity index \( I_p \) should be between 8% and 30%.

As natural sealing material, clay from the site close to the repository at the Mochové site was selected, the geo-technical parameters of which are tested in the laboratory.
2.2. Geotechnical parameters experimental measurement for selected sealing material

In cooperation with the Czech Technical University in Prague, rheological features of proposed sealing material by physical modelling are investigated. Repeated burdening with humid, freezing, and heating conditions in laboratory conditions has accelerated degradation processes of samples during the experiment. Degradation process of the compacted clay started on 12.2.2004. This experiment should confirm that degradation changes in material will not be such high, that material parameters do not fulfil initial requirements [1].

Procedure for acceleration of the degradation process of the clay material:
1. degradation cycle: drying (80 °C) - 1 week
2. degradation cycle: moistening (\( w = w_{optimal} \))
3. degradation cycle: freezing (–25 °C) - 1 week.

Experiment schedule:
1. phase - input tests – basic characteristic of the clay
2. phase - May 2004
3. phase - October 2004
4. phase - May 2005
5. phase - October 2005.

According to report [1] the following experiments will be performed until November 2005 in Czech Technical University laboratories:

- Basic characteristics: particle-size analysis, soil classification, Proctor standard experiment (compaction), density and optimal moisture
- Main parameters: hydraulic conductivity, liquidity limit, shrinkage limit, plasticity limit, plasticity index, cohesive shear stress, angle of friction, compressibility oedometric module, swelling stress, thermal conductivity, volumetric heat capacity.

In the following tables one can see results of experiments which were done during first three phases [2].

Table I. Basic characteristic of material for clay layer (1. phase)

<table>
<thead>
<tr>
<th>Property</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>soil density [kg/m³]</td>
<td>2 659</td>
</tr>
<tr>
<td>weight moisture [%]</td>
<td>16.0</td>
</tr>
<tr>
<td>soil classification</td>
<td>sandy clay with medium plasticity</td>
</tr>
<tr>
<td>Proctor standard experiment</td>
<td></td>
</tr>
<tr>
<td>( \rho_{d max} ) [kg/m³]</td>
<td>1 730</td>
</tr>
<tr>
<td>optimal moisture ( w_{opt} ) [%]</td>
<td>17.1</td>
</tr>
</tbody>
</table>

Table II: Physical modelling of periodical weathering influence to selected rheological characteristics of selected sealing natural material

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>1. phase</th>
<th>2. phase</th>
<th>3. phase</th>
</tr>
</thead>
<tbody>
<tr>
<td>hydraulic conductivity [m/s]</td>
<td>( 2.37 \cdot 10^{-12} )</td>
<td>in progress</td>
<td>in progress</td>
</tr>
<tr>
<td>swelling stress [kPa]</td>
<td>&lt; 20</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>angle of friction [°]</td>
<td>14°</td>
<td>14°</td>
<td>14°</td>
</tr>
<tr>
<td>cohesive shear stress [kPa]</td>
<td>228</td>
<td>110</td>
<td>97</td>
</tr>
</tbody>
</table>

Simultaneously under the same conditions, progressive developing of clay coherence is pursued on little physical model.
2.3. Construction of in-situ small-scale physical cover model

Before the covering model construction in 2005 the compacting test will be fulfilled to verify optimal technology for clay treating. In following figure you can see cross section of the in situ small-scale physical model of repository cover (55 x 55 m), which will be constructed at Mochovce site until November 2005 [3].

![Diagram of in-situ small-scale physical cover model](image)

**FIG. 1: Schematic cross section of the in situ model**

For long term monitoring (15-20 years) of clay sealing layer, the most significant component of final cover, in situ small-scale physical model of repository cover was designed [4]. Influence of weathering and settlement on geometrical shape of model, surface erosion, water infiltration and further selected parameters will be monitored. Along with sampling and laboratory measurements as well on-line monitoring of temperature and moisture content in cover model soils will go on.

The most significant indicator of damp-proof characteristic of covering material is soil moisture content. Therefore temperature and soil moisture content are monitored periodically every hour at 5 different depths in 40, 80, 120, 200 and 280 cm under surface. 6 main and 6 reserve monitoring points with 2 devices are placed at every depth. Sensors at 40 cm above the top monitors water infiltration in soil, devices at 80 cm indicate moisture content in overlapping soil. Monitoring at remaining depths indicates vertical profile of moisture content in clay layer. Output signals from all measurement equipment will be collected and processed through data-logger. Additionally every 4 years the sampling and laboratory measurement of soil moisture content, volume weight and permeability from four different bore holes will be prepared. Supporting information on site meteorological conditions as air and soil temperature and precipitation are measured at a nearby meteorological station.

From long-term point of view the evolution of surface environment under influence of vegetation, erosion and the animals is notable topic of monitoring. Alterations of surface conditions, shape of covering model and crop variety are periodically followed and undesirable changes are eliminated according the project. Furthermore influence of settlement or another natural issues on covering model shape stability is measured by standard geodetic methods at the end of every season.

2.4. Mathematical modelling of cover model reological characteristics

For the mathematical modelling of the cover model, the following six scenarios [5] were proposed:

- **normal evolution scenario for in-situ model:**
  - normal evolution scenario - without integrity failure (e.g. fracture), boundary conditions according to mean atmospheric conditions at Mochovce site;

- **climatic influence to sealing capability of cover layers:**
  - scenario for modelling of extremely low temperature influence,
  - scenario for modelling of very intensive rainfall influence,
  - scenario for modelling of erosion influence;
fracture influence to sealing capability of cover layers: scenario for modelling of vertical fracture influence

tenseness subsoil influence to sealing capability of cover layers
scenario for modelling of possibility of creation of fracture in cover layers.

The first five proposed scenarios will be modelled with the HYDRUS-2D code and the sixth scenario will be modelled with the PLAXIS code.

The HYDRUS-2D code [6] is a Microsoft Windows based modelling environment for analysis of water flow and solute transport in variably saturated porous media. The software package includes the two-dimensional finite element model HYDRUS2 for simulating the movement of water, heat, and multiple solutes in variably saturated media. The model includes a parameter optimisation algorithm for inverse estimation of a variety of soil hydraulic and/or solute transport parameters. The model is supported by an interactive graphics-based interface for data-pre-processing, generation of a structured mesh, and graphic presentation of the results. The HYDRUS2 program is a finite element model for simulating the movement of water, heat, and multiple solutes in variably saturated media. The program numerically solves the Richards' equation for saturated-unsaturated water flow and Fickian-based advection-dispersion equations for heat and solute transport.

PLAXIS code [7] is a range of finite element packages intended for 2D and 3D analysis of deformation, stability and groundwater flow in geo-technical engineering. Geo-technical applications require advanced constitutive models for the simulation of the non-linear and time-dependent behaviour of soils. In addition, since soil is a multi-phase material, special procedures are required to deal with hydrostatic and non-hydrostatic pore pressures in the soil.

3. Summary
The results of these research activities indicate the performance of the final covering of the repository at the Mochovce site.

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Study on a long-term alteration of engineered barrier materials

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Abstract. It is important to investigate the interaction between bentonite and cementitious materials in order to evaluate a long-term performance of engineered barrier system in the TRU waste repository. Alteration analysis of barrier materials was performed by using the calculation code (PhreeqC-Trans) coupled geochemical reaction and mass transport. Typical mineralogical alteration in engineered barriers (dissolution and formation of secondary minerals) and the change of physical properties (porosity, hydraulic conductivity and diffusion) was estimated from these preliminary calculations. Key technical issues on evaluation of alteration behaviour were clarified from a viewpoint of improvement of reliability on a long-term performance assessment. The experiments have been conducted to understand the alteration phenomena, which are predicted from the numerical analysis but have not experimentally confirmed. Furthermore, the ignored events of engineered barrier alteration on the modelling have been considered to evaluate the influence on the barrier performance.

1. Introduction

Cementitious material will be used for waste form, fillings, structure and lining in TRU waste repository. Bentonite material is also expected to be used for buffer material to restrict groundwater flow into the repository. Cementitious material is considered to induce the alkaline condition of the groundwater in the repository environment and these conditions may affect the performance of the buffer materials. Therefore, it is important to investigate the interaction between bentonite and cement in order to evaluate a long-term performance of engineered barrier system (EBS).

The performance assessment of EBS had been conservatively evaluated because of the uncertainties of the long-term alteration behaviour [1]. However, more realistic evaluation might be possible by accumulation of recent experimental knowledge on engineered barrier alteration in the repository environment and by development of modelling and numerical analysis method on the alteration process.

In this study, key technical issues on evaluation of a long-term alteration of EBS were clarified from the results of preliminary numerical analysis. In order to solve these technical issues, some experiments have been conducted to understand the alteration phenomena, which are predicted from the numerical analysis but have not experimentally validated. Furthermore, the ignored events of engineered barrier alteration on the modelling have been considered to evaluate the influence on the barrier performance.

2. Alteration analysis and identification of technical issues

2.1. Alteration analysis of EBS

A long-term alteration of the EBS in the TRU waste repository was evaluated based on the recent knowledge about geochemical and hydrological processes. Coupled geochemistry and mass transport calculation code (PhreeqC-Trans) was used in the preliminary numerical analysis. This code considers the geochemical reactions and the mass transport (i.e. diffusion and advection). The hydraulic conductivities, the diffusion constants and the porosities were renewed in accordance with the change of mineralogy in the EBS, and the mass transport parameters were recalculated at regular time intervals.

The geometrical configuration for the numerical analysis is shown in Figure 1. TRU waste repository system consists of waste package, filling mortar, concrete structure, buffer material (bentonite/sand
mixture), concrete lining and rock. Representative section of this system was selected and modeled as 1 dimensional geometry. In the alteration analysis, ordinary portland cement and Kunigel V1® (Japanese bentonite) were considered as cementitious and bentonite materials, respectively.

**FIG. 1: Geometrical configuration for the numerical analysis**

Figure 2 indicates a calculated result of the typical alteration behaviour in the engineered barriers. Dissolution of the chalcedony was predicted in the buffer material (bentonite/sand mixture). Most of the montmorillonite was altered to the analcime at the interface between bentonite/sand mixture and concrete structure. The portlandite in cementitious materials dissolved to form the CSH with a low Ca/Si molar ratio due to the reaction with silica ions dissolved from the chalcedony. As the result of these reactions, the porosity increased in the bentonite/sand mixture. On the other hand, the porosity at the interface between concrete structure and bentonite/sand mixture significantly decreased by the precipitation of the CSH. Furthermore, the change of physical properties (i.e. hydraulic conductivity and diffusion) was estimated in accordance with the mineralogical alteration of the EBS.

**FIG. 2: Typical alteration behaviour and porosity change in the engineered barriers predicted from numerical analysis (Groundwater type: FRHP [2])**

2.2. Identification of key technical issues

Through the various preliminary analyses, the following key technical issues were clarified from a view point of improvement of reliability on a long-term performance assessment.
3. Experimental and analytical studies to solve the technical issues

3.1. Experimental validation of alteration behaviour

Some experiments were planned and have been continued to understand the alteration phenomena of barrier materials, which are predicted from the preliminary alteration analysis but have not experimentally confirmed. These experimental results will be compared with the calculated results to discuss the applicability of alteration model and to improve the reliability on a long-term performance assessment.

(a) Cementitious material:
- Applicability of alteration model to the realistic cement (Fly ash cement etc.) and change of physical and chemical properties of cementitious material with a crack;

(b) Bentonite material:
- Reversible alteration reaction of analcime to montmorillonite under the condition of comparatively low pH and low Na concentration (i.e. Expected condition of cement degradation in Region III) and influence of accessory minerals and saline water on the alteration behaviour of montmorillonite;

(c) Cement/bentonite interaction:
- Identification of secondary minerals at the bentonite/cement boundary and development of separation technique to identify a small amount of secondary minerals

3.2. Consideration of ignored events on the alteration model

Some of alteration events, assumed not to have critical influence on the barrier performance from the result of a short-term experimental evaluation, had been ignored in the alteration analysis. However, in order to improve the reliability on a long-term performance assessment of EBS, it is necessary to consider these ignored events of engineered barrier alteration on the modelling. The influence of aggregates in cementitious material and mixed silica sands in bentonite material, which are assumed as non-reactive materials, is considered on the numerical analysis to evaluate one of these ignored events.

Figure 3 shows the calculated result considering the influence of aggregates and mixed sands. In both case (aggregates and mixed sands are considered as non-reactive or reactive materials), dissolution of chalcedony in the bentonite/sand mixture and precipitation of CSH in mortar was the common alteration behaviour. On the other hand, remarkable precipitation of CASH was estimated in mortar due to the dissolution of anorthite when aggregates and mixed sands were considered as reactive materials.

The residual ratio of minerals to initial aggregates was indicated in Figure 4. Sufficient amount of anorthite and albite was estimated to dissolve to form secondary minerals in mortar although the other minerals in aggregates such as quartz were not affected to the alteration behaviour. The difference of residual ratio of minerals is caused by the dissolution rate of each mineral. From these analytical results, it is suggested that comparatively reactive minerals in aggregates or mixed silica sands should be considered for the reliable evaluation of a long-term alteration behaviour.
Sands and aggregates are considered as non-reactive materials

Sands and aggregates are considered as reactive materials

Volume fraction of minerals in the barrier materials after 5,000 years

FIG. 3. Influence of aggregates in cementitious material and mixed sands in bentonite material on the alteration behaviour of barrier materials

Mortar

Residual ratio of minerals in aggregates (Mortar part)

FIG. 4. Residual ratio of minerals in aggregates (Mortar part)

Acknowledgement. This study has been funded by the Japanese Ministry of Economy, Trade and industry (METI).

REFERENCES

Abstract. This work describes the regulatory approach to safety evaluation of radioactive waste operations facilities in Australia. Australian Radiation Protection and Nuclear Safety Agency (ARPANSA) is the regulatory authority for commonwealth entities operating nuclear installations including Radioactive Waste Operations facilities. In assessing the application for operating nuclear installations ARPANSA assessor prepares a safety evaluation report which is a recommendation to the Chief Executive Officer (CEO) of ARPANSA whether to issue a licence. For the assessment purpose the operating organisation requires addressing certain matters to demonstrate that the safe operation of the facility. The requirements are set out in the Australian Radiation Protection and Nuclear Safety Act 1998 (the Act) and the Australian Radiation Protection and Nuclear Safety Regulations 1999 (the Regulations) as amended. In addition, ARPANSA regulatory Guideline and Regulatory Assessment Principles are applied in the safety evaluation. The safety evaluation report resulted in a recommendation to the CEO to issue a licence to Australian Nuclear Science and Technology Organisation (ANSTO) to operate Waste Operations facilities. The authorisation of operation was subject to compliance with certain standard and special licence conditions. The assessment resulted in eight special licence conditions, which were specific to Radioactive Waste Operations facilities in the areas of safety management, criticality safety and emergency arrangements.

1. Introduction

In compliance with the Act and the Regulations ANSTO, a commonwealth entity, applied to ARPANSA for a licence to operate the facilities of Waste Operations.

The facilities are nuclear installations in accordance with the Act. The facilities include:

- Solid waste facilities including storage of miscellaneous intermediate level solid waste, storage of drummed predominantly low level solid waste, storage of safeguards and other nuclear materials, storage of sealed sources.
- Liquid waste facilities including the effluent collection treatment system, miscellaneous treatment and storage of low level and intermediate level liquid waste.
- Laundry and decontamination facilities, low-level solid waste processing and interim storage of solid waste.
- Intermediate level liquid waste storage facility for receipt, storage and retrieved for processing of intermediate level liquid waste.

The application was assessed applying appropriate regulatory tools including the Act and the Regulations, Regulatory Guideline and Regulatory Assessment Principles, international best practice, safety analysis report to determine the efficacy of safety provisions of the facilities and their conformity to regulatory requirements. Details of the matters considered in the safety evaluation are presented in the following sections.

2. Elements of assessment

In assessing the application for operating licence ARPANSA took into account the description of the facility, plans and arrangements for managing safety, technical specifications (including standards and codes used), safety analysis and operational limits and conditions.

The plans and arrangements for managing safety incorporate arrangements for maintaining effective control, quality system, safety management plan, radiation protection plan, radioactive waste
management plan, security plan and emergency plan. The plans and arrangements for managing safety are assessed against the set of Regulatory Guidelines [1]. Such guidelines are based on international best practice in radiation protection and nuclear safety, drawing from national and international publications and experience, especially from the International Atomic Energy Agency (IAEA).

The safety of ANSTO Waste Operations facilities was assessed utilising Regulatory Assessment Principles [2] which is based on international publications and experience, especially from International Nuclear Safety Advisory Group (INSAG) and the IAEA. For each of the principle stages in the life of the nuclear facilities ARPANSA required that the operating organisation submit an updated safety case. The safety case demonstrates that throughout its life, the facility complies with the radiation dose limits specified in the Regulatory Assessment Principles [2]. The safety case includes the design information for the facility, the operational limits and conditions (OLCs) within which the facility must operate, and a safety analysis that is documented in a safety analysis report (SAR).

3. Safety analysis report for radioactive waste operations facilities


ANSTO carried out risk ranking studies to identify potential hazards in order to address the safety requirements associated with the waste management process. The actions or recommendations originated from the risk and hazard analyses were implemented prior to finalisation of the safety analysis report and submission to ARPANSA.

The ANSTO Waste Operations SAR also described the general aspects of operation including overview of the process, the scope, the interfaces and the responsibilities of the operators, the supervisor and the facility officer.

General safety principles of the design and development were discussed as well as hazards such as nuclear, radiological, chemical, industrial protection systems, fire alarms and suppression, process alarms and security. The policy and procedures for safety management, waste management and review of the operating experience and plant condition were also included in the SAR.

The safety analysis section is a major section of the SAR. The section describes the risk management based on the identified risks, safety analysis for normal operation, abnormal events and external events, analysis of environmental impact, assessment of operating limits and conditions and the issues related to occupational health and safety.

ARPANSA utilised safety principles described in the Regulatory Assessment Principles [2] to assess the safety aspects described in the SAR. ARPANSA also applied the defence in depth concept for assessing the safety of ANSTO Waste Operations facilities. It was found that the technologies incorporated in the design of the facilities and associated handling equipment were proven adequately safe.

4. Results of assessment

After assessing all relevant information ARPANSA assessor prepared a Safety Evaluation Report (Regulatory Assessment Report) [4], which is a recommendation to the CEO of ARPANSA whether to issue a licence. This report was based on the results of the detailed assessment of the application and the resolution of issues resulted from any public submission on the ANSTO Waste Operations application. The Safety Evaluation Report is a complete assessment of the application for an Authorisation to operate ANSTO Waste Operations Facility. This report demonstrates that the conduct for which the licence is sought can be effectively controlled to provide adequate protection to the health and safety of the people and the environment. The issues which are not clearly addressed were subjected to special licence conditions. The assessment resulted in eight special licence conditions in the areas of safety management, criticality safety and emergency arrangements.
5. Conclusion
The matters considered in issuing authorisation to operate ANSTO Waste Operations facilities include general information, plans and arrangements for managing safety, final safety analysis report, operational limits and conditions, arrangements for commissioning and operation, undue risk to the health and safety of the people, and to the environment, net benefit from the conduct, ALARA doses, ability to comply with the regulations and licence conditions and the content of any public submission.

The safety evaluation suggested that the design of ANSTO Waste Operations facilities is acceptably safe for the period of intended operation and ARPANSA regulatory system provides an effective framework to ensure that the ANSTO Waste Operations facilities are operated in a safe manner.

REFERENCES
The technical issues of seismic and volcanic activity for the safety assessment of high-level radioactive waste disposal in Japan

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Abstract. There is volcanic and seismic activity in and around the Japanese Islands. Based on Feature Events Processes (FEPs) of OECD/NEA International Database, geological scenarios affecting a high-level radioactive disposal system domain in Japan are identified for the safety assessment. Some scenarios can be avoided at a repository by means of site selection, whereas others are unavoidable during the siting process. Each scenario contains geological uncertainties. These uncertainties are technical issues in geosciences. Further research on seismic and volcanic activity is necessary to reduce the remaining uncertainty of nuclear waste disposal safety assessment in Japan.

1. Introduction

Most European countries are located on stable areas without volcanic eruptions and great seismic events, whereas Japan is on a mobile belt accompanied by intensive volcanic and seismic activity. These natural phenomena are caused by the present plate movement. Japanese Islands are located along convergent plate boundaries where the Pacific and Philippine Sea Plates are subducting under the North American and Eurasian Plates.

In case of sudden natural phenomena, such as earthquakes and volcanic eruptions, the influence on the deep geological environment could be substantial; however disruptive areas are relatively restricted in extent. On the contrary, uplift, subsidence, denudation, climate and sea level change are gradual and regional [1]. Based on the results of previous geological studies, it is possible to select a sufficiently stable geological environment in Japan, such that a repository will not be influenced by disruptive natural phenomena for at least the next hundred thousand years [2].

2. Requirement of geological environment

The Final Disposal Act was enacted in 2000 to ensure systematic and reliable disposal. The investigation should be done in three steps, that is, Literature Surveys, Preliminary Investigation, and Detailed Investigation. Legal requirements to select Preliminary Investigation Areas are 1) no record of significant movement in geological formations due to earthquake or fault activity, igneous activity, uplift, erosion and other natural phenomena, 2) small possibility of significant movement in the future due to earthquake or fault activity, igneous activity, uplift, erosion and other natural phenomena, 3) no record of unconsolidated deposits from the Quaternary period, and 4) no record of mineral resources that are economically valuable.

In 2002, the Nuclear Safety Commission presented the requirements for a geological environment obviously improper to a repository site [3]. It includes 1) uplift/subsidence/erosion, 2) earthquake/fault movement, 3) volcanic/igneous activity, 4) mineral resources, and 5) bedrock. The major effects of large earthquake/fault movements to a repository and its geological environment are 1) direct breakage of repository and waste accompanied by rupture and crush of bedrock, and 2) change of groundwater flow and chemistry due to the formation of major groundwater flow paths accompanied by the rupture and crush of bedrock, and groundwater pressure change caused by bedrock strain. The major effects of volcanic/igneous activity to a repository and its geological environment are 1) direct breakage of repository and waste accompanied by intrusion and/or eruption of magma, and 2) increase of geothermal gradient and development of thermal convection caused by magmatic heat, and change of groundwater chemistry due to the mixing with geothermal and volcanic gases.
3. **FEP of geological and climatic processes end effects**

The international database of FEPs list from the OECD/NEA [4] covers the items necessary for the safety assessment of radioactive waste disposal. We analyse the FEPs to determine the relationships between individual FEPs in the area of geological and climatic processes and their effects [5, 6]. The Analyses were done for items 1.2 (geological processes and effects) and 1.3 (climatic processes and effects) in Chapter 1 (external factors). First, we screened FEPs in light of the geological setting of Japanese Islands. For example, seismicity (1.2.03) and volcanic and magmatic activity (1.2.04) are essential in Japan, but glacial and ice sheet effects (1.3.05) is less considerable because the land has not been covered by an ice sheet during the Quaternary. Then we sub-categorized major FEPs to fit the geological situation in Japan. For example, the seismicity FEP can be subcategorized into large earthquake along active fault in the land region, along the subducting plate boundary, and along the collision plate boundary; shallow and large earthquake in land region where any active faults have not been discovered; liquefaction, mud volcano and landslide triggered by earthquake. The volcanic and magmatic activity (1.2.04) can also be sub-categorized as intrusion, eruption, and caldera-forming large-scale eruption in their activity, and as existing volcano and newly-formed volcano in their time of formation. We also specified effects for each FEP. Finally we identified 57 scenarios affecting a disposal system domain, such as “tectonic movement and orogeny (1.2.01) >> volcanic and magmatic activity (newly-formed volcano) (1.2.04) >> hydro-thermal activity (1.2.06) >> hydrological / hydro-geological response to geological change (1.2.10) >> disposal system domain”.

4. **Technical issues on seismic activity**

So many earthquakes occur in and around the Japanese Islands. Large earthquakes (M8-6) often occur along the subduction plate boundaries. Large earthquakes also occur along active faults running in the land region.

As the result of above mentioned FEP analyses, we identified 24 scenarios relating seismicity affects a disposal system domain. Thirteen scenarios can be avoided at a repository site by means of site selection. For example, a repository can be emplaced away from a known active fault running in the land region. The remaining technical issues in this case are the future migration of the fault and the extent of its area of influence.

On the contrary, the other 11 scenarios cannot be avoided during the siting process. For example, it is well known that groundwater level changes when a large earthquake occurs [7]. This process can be represented by the scenario of “tectonic movement and orogeny (1.2.01) >> seismicity (earthquake along the subducting plate boundary) (1.2.03) >> deformation (1.2.02) >> hydrological / hydrogeological response to geological change (1.2.10) >> disposal system domain”. The technical issue for this scenario is the assessment of groundwater movement accompanied with a large earthquake. The groundwater monitoring system for the future earthquake activity currently working in Japan could be used for studying the hydro-geological environment of a radioactive waste repository.

Another example among the 11 scenarios unavoidable during the siting process is a shallow and large earthquake in the land region where any active faults have not been discovered. Nation-wide active fault maps are available in Japan [8], but we do not have a complete list of all the active faults running in Japan. For example, in 2000 the Tottori-ken Seibu earthquake of M 7.3 occurred in an area where nobody had recognized a running active fault [9]. The trench survey conducted after the earthquake revealed the previous displacement along the fault zone. The recurrence interval of this earthquake was estimated to be between 10,000 and 40,000 years. This case study shows that there are active faults with low activity, which have not yet been identified in the Japanese Islands.

5. **Technical issues on volcanic activity**

A lot of Quaternary volcanoes are distributed in the Japanese Islands [10]. There is some regularity in their distribution. Volcanoes cannot be found on the Pacific Ocean side of the volcanic front in Northeast Japan. On the contrary, there is not a clear volcanic front in Southwest Japan, where monogenic volcanoes are sporadically distributed.
As the result of above mentioned FEP analyses, we identified 14 scenarios related to volcanic and magmatic activity affecting a disposal system domain. Six scenarios can be avoided at a repository site by means of site selection. For example, the repository can be placed away from a known Quaternary volcano. The remaining technical issues in this case are the future evolution of the volcano and the extent of its area of influence.

The other 8 scenarios cannot be avoided during the siting process. For example, caldera forming during a large-scale eruption of a Quaternary volcano affects huge areas in Japan. Previous records of caldera-forming large-scale eruption in Kyushu shows pyroclastic flow deposits over large areas, and thick piles of volcanic ash all over the Japanese Islands. This is not only a problem concerning nuclear waste repository.

Another example among the 8 scenarios unavoidable during the siting process is a newly formed volcano. A new volcano away from the volcanic front and other Quaternary volcanoes is a highly technical issue with regards to geology and geophysics. As magma is derived from lower crust and upper mantle, there is a thermal anomaly under the volcanic area, where the velocity of seismic waves are low. Anomalous mantle, which can be detected by a geophysical method, could be identified under a potential area of new volcano formation.

6. Conclusion
Geological scenarios affecting the disposal system domain have been identified for the safety assessment. Each scenario contains geological uncertainties. These uncertainties are technical issues in geosciences. Further research on geological processes is necessary to reduce the remaining uncertainty of nuclear waste disposal safety assessment in Japan.

REFERENCES

Challenges affecting the implementation of waste management policies and strategies in Zambia.

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Abstract: This brief presentation gives some of the highlights of the types of radioactive waste that is in the country. Due to scanty information available for the radioactivity in most of the tailing dumps and waste rock in most of the mining areas, the paper has concentrated on the sealed radiation waste that has been generated over the years in the Zambian industries, medical and research institutions. Challenges and recommendations that should be taken to ensure adequate protection of the public are discussed.

1. Introduction:
Mining in Zambia has for many years been of special interest to the Zambian people. Currently it accounts for 70% of our economic dependence, 25% of our employment and 13% of our gross domestic product. The Zambian government expects copper production to rise to about 750,000 tonnes per annum with coming on board of Luanshya in the copperbelt province, Lumwana and Kansashi copper mines in the North western province in Zambia.

Until recently, occurrences of uranium in Katanga system were known only in the Shaba province of the Democratic Republic of Congo and on the Copperbelt province in Zambia. But a survey by AGIP has shown the existence of uranium mineralization in the north-western province in Zambia. The information on the radioactivity of all the tailing dumps in the country and waste rock is scanty. However, it suffices to indicate that uranium is not a mineral of major interest to the country but the major mining activities are being done in an area that has uranium mineralization. In all the mines in Zambia, tailings and waste rock have been disposed of close to the mine sites.

The safe management of sealed radioactive waste and specifically the need to protect humans and the environment now and in the future has been given particular attention by the Zambian government. Sealed radioactive waste has been generated in industry, medical and research institutions and includes among others Co-60, Cs-137, Ir–192 and Ra-226 needles which were conditioned with the help from the International Atomic Energy Agency (IAEA).

2. Legal framework
There are two main pieces of legislation that directly deal with the protection of humans and the environment from the harmful effects of ionizing radiation. The Ionizing Radiation Act, Chapter 311 of Laws of Zambia which establishes the Radiation Protection Board (RPB) as the national competent authority and also prescribes its functions and powers to enable RPB discharge its functions. On the other hand, there is the Environmental Protection and Pollution Act that provides for the protection of the environment and control of pollution and establishes the Environmental Council of Zambia and prescribes the functions and powers of the Council.

3. Disposal facilities
There is no site in the country that has been approved as a disposal site for any form of radioactive waste. The sealed radioactive waste that has been generated over the years is currently being stored in Kalulushi Interim Storage Facility. The activities at this storage facility have mainly concentrated on the safety and security of sealed radioactive waste. An inventory of disused and conditioned radioactive sources stored in the facility is available.
4. Challenges affecting the implementation of waste management programme

4.1. Lack of a national policy and strategy for waste management

The application of nuclear science and technology was introduced in the country long before the enactment of the Ionizing Radiation Act, Chapter 311 of the Laws of Zambia. The importation of the sources by the various users in industry, medical and research institutions in the past was done without entering into contractual agreements for the radiation sources to be returned to the suppliers upon becoming obsolete. These sealed radioactive waste sources together with the “orphaned” sources collected by government security agencies from the public domain have to be managed for a considerable amount of time before the radioactivity decays to levels below the approved clearance limits.

The privatization process of most industries in the country especially the mining sector has led to a situation where the new investors have not fully accepted the liabilities associated with the management of obsolete radiation sources. The Zambia Consolidated Copper Mines-Investment Holding (ZCCM-IH), an entity that government has created to take care of historic liabilities arising from the privatization process of state owned facilities has ensured the safety and security of the obsolete sources from the mining sector. However, there is need to have a clear policy and strategy in place at a national level with a specific institution being responsible for waste management. The ZCCM-IH is an entity that might not exist in perpetuity and hence the need for government to urgently address the issue of waste management at a national level.

4.2. Inadequacies in the legal framework

An attempt has recently been made to ensure that the present legislative framework pertaining to radiation protection and waste management is made consistent with the principal requirements of the basic safety standards. However, currently there is more attention paid to the management of sealed radioactive waste than the mining waste (tailings and waste rock). The issue of technologically enhanced natural radioactivity requires to be covered by law. There is an urgent need to ensure that a comprehensive environmental impact assessment is done prior to any development. This assessment should include among other requirements a comprehensive radiological hazard assessment. In the period of 1957-1990 uranium was exploited at Mindolo Shaft 4 in Kitwe and the tailings dump are within the vicinity of a village. Further, waste rock dumps at Luanshaya (Ore Body 2 and the No. 28 Shaft Dump) and at Nchanga (Ore body 2) contain radioactive rock fragments however, in Kitwe, waste rock from Mindolo Shaft 4 is occasionally used for road and house construction.

4.3. Inadequate waste disposal facilities in the region

The closest radiation waste disposal facility in the sub-region is in South Africa. However, accessibility of this radiation waste facility in South Africa by neighboring countries could be difficulty because of the complexity of the issue and political and social concerns.

4.4. Recommendations

There is need for international organizations like the International Atomic Energy Agency to assist member states especially those in developing countries to develop at a national level a radiation protection and waste management policy and strategies in order to upgrade safety of the public and environment from harmful effects of ionizing radiation.

Harmonization of national pieces of legislation is very cardinal if adequate protection of both humans and environment is to be provided. Laws that relate to environment, ionizing radiation protection, and mining should have adequate provisions for all practices that have a potential of resulting in undue exposure to sources of ionizing radiation. Countries where uranium is not a major element of interest but mining is done in an area with known uranium deposit should be assisted to ensure that tailings and waste rocks with high radioactivity are not accessible by the public and the industrial use of such waste should not pose a radiological health hazard to humans and environment.

At an international and/or regional level, a mechanism should be established probably with the help of regional and/or international bodies so that countries that have a well-developed technological base
and have radiation waste facilities that are operational should consider accepting some radiation waste especially sealed radiation sources from developing countries like Zambia.

5. Conclusion

The issue of sealed radioactive waste disposal is currently a big problem in Zambia as the choice of an appropriate disposal option is yet to be made. Recently imported sealed radiation sources will not be difficult to dispose of as they will be required to be returned to the supplier for disposal. There is now a regulatory requirement so as to minimize the generation of radioactive waste. However, the sources that were imported prior to this regulatory requirement will need to be managed for a considerable amount of time before they can be removed from regulatory control. In addition, appropriate interventions are required to deal with the radiological health hazard associated with tailing dumps and waste rock that come from mines that have been considered to have above normal radioactivity.
Radioactive waste management strategy in Lithuania

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Abstract. After the decision to close Unit 1 of the Ignalina Nuclear Power Plant in 2004, it became clear that during decommissioning a large amount of radioactive waste would arise, which will need to be managed properly together with already existing spent fuel and future spent nuclear fuel. All this meant a need for improving the infrastructure of radioactive waste management which had resulted from adaptation, by the Lithuanian Government, of a radioactive waste management strategy for the first time in 2002. The basic objectives that shall be implemented in Lithuania with the purpose to improve the management of radioactive waste, including disposal options, are given in the strategy. This elaborates directions of the management of solid and liquid radioactive waste of the Ignalina NPP, management of spent fuel, management of radioactive waste generated by small producers, together with necessary procedures, and introduces necessary systems and directions for scientific research.

1. Introduction

In Lithuania, the State Enterprise Ignalina Nuclear Power Plant (INPP) is the biggest producer of radioactive waste and the only producer of spent fuel. Hospitals, scientific institutions and industry produce only a minor amount of radioactive waste. Therefore the respective Lithuanian policies and radioactive waste management strategy are oriented mainly to the INPP.

On legal basis, all radioactive waste management facilities, storages and repositories are considered to be nuclear energy facilities; consequently the same safety requirements and licencing system are applied to such kind of facilities as for nuclear energy activities.

In 1990 Lithuania re-established independence and took over the management of nuclear energy in Lithuania from Russia. In 1993 Lithuania became a Member State of the IAEA and started improving safety in the nuclear sector including safety in radioactive waste management. At present, the management of the waste meets partly international requirements and there exists a need in Lithuania to improve the management of radioactive waste.

2. Radioactive waste management strategy and policies

2.1. Radioactive waste management strategy

After the decision to close Unit 1 of the Ignalina Nuclear Power Plant in 2004, it became clear that during decommissioning a big amount of radioactive waste would arise, and that it should be managed in a proper way together with already existing spent fuel and future waste and spent nuclear fuel.

A Radioactive Waste Management Strategy was approved by the Government of the Republic of Lithuania for the first time in 2002. The strategy will be renewed after 5 years (in 2007). The Strategy sets main objectives that should be implemented in Lithuania in order to improve the radioactive waste management:

- to strive for implementation of a proper radioactive waste management policy;
- to develop the radioactive waste management infrastructure based on modern technologies;
- to create the effective financing system for radioactive waste management;
- to provide for the set of practical actions that shall bring management of radioactive waste in Lithuania in compliance with the radioactive waste management principles of IAEA and with the good practices in force in EU Member States.
Tasks of the strategy are:
- to improve the legal basis for radioactive waste management;
- to modernize a system of radioactive waste management at the Ignalina NPP and to implement the new radioactive waste classification system;
- to provide INPP with necessary facilities for radioactive waste management arising from decommissioning;
- to modernize the management infrastructure for radioactive waste generated by small producers;
- to implement new repositories for radioactive waste.

The strategy elaborates directions of management of solid and liquid radioactive waste of the Ignalina NPP, management of spent fuel, management of radioactive waste generated by small producers, also necessary procedures, introducing necessary systems and directions for scientific research.

### 2.2. Ignalina NPP solid waste management

At the Ignalina NPP, management of the radioactive waste should be re-organized. For that the best technologies should be implemented which would allow to reduce the general activity and volume of the waste. At present, all radioactive waste is stored on site in concrete storages from the beginning of INPP operation. To meet modern requirements of waste storage all solid radioactive waste should be retrieved from existing facilities, characterized, conditioned and solid short-lived radioactive waste should be sent back to storage facilities. Long-lived waste should undergo proper treatment and be stored at the long-live radioactive waste storage facility without final immobilization until final management methods would be decided upon.

At the Ignalina NPP the operational wastes have indeed been accumulated and are still being accumulated in existing storage facilities on the basis of an old classification:

<table>
<thead>
<tr>
<th>Waste Group</th>
<th>Surface dose rate (at 10 cm)</th>
<th>Specific activity</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>mSv/h</td>
<td>β-activity</td>
</tr>
<tr>
<td>Low Level Waste</td>
<td>1.0×10⁴ – 0.3</td>
<td>7.4×10⁴ - 3.7×10⁶</td>
</tr>
<tr>
<td>Intermediate Level Waste</td>
<td>0.3 – 10</td>
<td>3.7×10⁶ - 3.7×10⁹</td>
</tr>
<tr>
<td>High Level Waste</td>
<td>&gt; 10</td>
<td>&gt; 3.7×10⁹</td>
</tr>
</tbody>
</table>

The new radiological classification of solid waste, which had been issued by the State Nuclear Power Safety Inspectorate in 2001, will be used for decommissioning waste and for the waste disposal streams.

The Ignalina NPP is implementing a project on designing and establishing a solid radioactive waste processing and storage complex near INPP, which will allow the plant to have modern equipment to manage existing solid waste which arose during operation, and solid waste to be generated during decommissioning. An International Ignalina Decommissioning Support Fund finances the project.

Both reactors of INPP contain more that 3700 t of graphite. Possible ways of the graphite management were analyzed in the INPP Final Decommissioning Plan. However no one option was chosen yet as the most suitable, and the decision on the way of managing the graphite is not yet taken.
Table 2. New radiological classification of solid radioactive waste.

<table>
<thead>
<tr>
<th>Waste Group</th>
<th>Definition</th>
<th>Surface dose rate</th>
<th>Conditioning</th>
<th>Disposal method</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>Exempt waste</td>
<td>Not required</td>
<td></td>
<td>Management and disposal as usual conventional waste</td>
</tr>
<tr>
<td>A</td>
<td>Very low level waste</td>
<td>≤0.5 mSv/h</td>
<td>Not required</td>
<td>Very low level waste repository (Landfill)</td>
</tr>
<tr>
<td>B</td>
<td>Low level waste</td>
<td>0.5-2 mSv/h</td>
<td>Required</td>
<td>Near surface repository</td>
</tr>
<tr>
<td>C</td>
<td>Intermediate level waste</td>
<td>&gt;2 mSv/h</td>
<td>Required</td>
<td>Near surface repository</td>
</tr>
<tr>
<td>D</td>
<td>Low level waste</td>
<td>≤10 mSv/h</td>
<td>Required</td>
<td>Near surface repository (cavities at intermediate depth)</td>
</tr>
<tr>
<td>E</td>
<td>Intermediate level waste</td>
<td>&gt;10 mSv/h</td>
<td>Required</td>
<td>Deep geological repository</td>
</tr>
<tr>
<td>F</td>
<td>Spent sealed sources</td>
<td></td>
<td>Required</td>
<td>Near surface or deep geological repository</td>
</tr>
</tbody>
</table>

2.3. Ignalina NPP liquid waste management

Liquids, spent resins, concentrated sludge should be cemented and packed in packages suitable for transportation, storage and disposal. In 2005, a new liquid waste cementation facility will start operation. Liquid waste will be cemented in 200 l drums and then packed in concrete containers and stored on site while disposal storage will be available.

Evaporators’ concentrates are bituminized. INPP performs a study on the possibility to convert existing bituminized waste storage to repository. Depending on the results of the studies, an appropriate decision will be taken: the bitumen radioactive waste storage facility should be licensed as a repository or the bitumen technology should be transformed in such way that waste forms are enclosed into suitable containers as required for storage, transportation and disposal in the near surface repository.

The possible ways of spent oil management were analyzed in the INPP Final Decommissioning Plan. However a decision on the way of spent oil management has not yet been taken.

2.4. Ignalina NPP spent fuel management

Spent fuel is stored at water pools near reactors and in a dry storage facility on site. Since 1999, spent fuel, after being stored in ponds near reactors, was loaded into CASTOR RBMK-1500 and CONSTOR RBMK-1500 casks and transported to the dry storage facility. Spent fuel can be stored in the casks for at least 50 years. However the storage is almost used up and INPP demands new storage.

In the frame of decommissioning INPP, the Ignalina NPP is implementing a project on designing and establishing a new spent fuel storage facility near INPP, which will allow the plant to store all present spent fuel accumulated in pools, and also spent fuel which will be produced during operation of Unit 2 (approx. 19000 fuel assemblies). The International Ignalina Decommissioning Support Fund finances the project.

2.5. Management of radioactive waste generated by small producers

Disused radioactive sealed sources are stored together with INPP radioactive waste on the site storages of INPP. In the Radioactive Waste Management Strategy it is foreseen that disused sealed sources shall be managed separately from other radioactive waste. Those sources which cannot be reused or
send back to a supplier shall be treated without final immobilization until the acceptance criteria of a deep geological repository will be set.

Existing “Radon” type radioactive waste storage facility shall be licensed as nuclear facility (the storage was closed in 1988). During the last years, physical protection of the site was improved. In the current year, the safety analysis will be finalized and a decision on its future should be taken: either to convert it into a repository or to retrieve the waste.

3. Final disposal of radioactive waste

As the Ignalina NPP is the main source of radioactive waste, the main research work to find sites for the final disposal of waste is carried in the vicinity of INPP: short distance from INPP (less transportation costs), low population density, and low land economic potential and good level of geological characterization are the main positive features of the INPP region. Also by the categorization of the land of Lithuania, that region is considered as a region of energy, industry and transport area. That fact allows to meet legal and environmental requirements, including more positive public acceptance.

Due to regional policy in Lithuania, search for sites with most suitable geological characteristics in the whole territory of the country has difficulties. Local municipalities have a significant influence on the final decision and are involved in the adoption of all documents of planning, designing and construction of nuclear facilities.

3.1. Very low activity radioactive waste

The Lithuanian Regulation covers unconditional free release limits. The unconditional free release limits apply for the clearance of materials, the destination of which is not defined “a-priori”. There is thus neither traceability nor regulatory control of the waste complying with the unconditional free release limits. The hypothetical critical individual affective dose resulting from the combination of all possible exposure pathways may not exceed 10 µSv/y. This very conservative approach results in low residual maximum allowable specific activities in the waste coming into consideration for unconditional free release. For the implementation of the conditional free release procedure, additional investigations should be undertaken.

For the other radioactive waste of very low activity, which exceeds free release limits, Ignalina NPP is preparing a project of designing and constructing a landfill repository, which will be commissioned in 2007. The capacity of the repository will be 60000 m³.

3.2. Near surface repository

Immobilized short-lived low and intermediate level waste will be transferred to the repository. In 2003, acceptance criteria for near surface disposal were set. According to the concept, the repository would consist of vaults with a total disposal volume of 100000 m³.

The Radioactive Waste Management Agency performed an area survey for the repository in the vicinity of the Ignalina NPP. One place with the best geological characteristics was chosen. A comprehensive environmental impact assessment for the site was finished in 2004.

The final decision for the site of the repository should be based on the requirements set forth by the Lithuanian laws, taking into account agreement from the local municipality, public acceptance and the opinion of the boarding country Belarus.

3.3. Long-lived radioactive waste and disused spent sources

Long-lived radioactive waste, spent fuel and disused spent sources can be disposed in a deep geological repository. The Lithuanian strategy for the disposal of that kind of waste is:

− to analyze possibilities to have a deep geological repository in Lithuania (to be sure whether suitable geological layers exists in the territory of Lithuania);
− to analyze possibilities to create a regional repository taking joint efforts of a few countries;
− to analyze possibilities for disposing spent fuel of in other countries and to find out on which estimated costs such disposal would be justified.
No clear direction for the disposal of long-lived radioactive waste or spent fuel is chosen at the present time. The fact that after closure of the Ignalina NPP no more spent fuel and big amounts of long-lived waste would be produced should also be taken into account. Presently packed spent fuel can be stored on site in dry storage for 50 or more years.

The Radioactive Waste Management Agency performs research of the geological layers in Lithuania. The first study on granite layers suitable for the deep geological repository, antagonized the public, politicians and media for the availability to dispose spent fuel in Lithuania.

4. **Conclusions**

1. Lithuania has a clear strategy for the improvement of the management of radioactive waste.
2. Ignalina NPP’s operational waste needs retrieval, reclassification and conditioning. New facilities will be installed.
3. A landfill and near surface repository will be commissioned in this decade.
4. The question on disposal of spent fuel and long-lived radioactive waste is still open.
Knowledge representation in HLW disposal facility safety assessment

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Abstract. A good understanding, by the public and licensing authorities, of the decision making process of radioactive waste disposal facilities safety assessment is crucial for confidence building and further discussions of the results. To this end, uncertainty analysis plays a very important role in such a multi and interdisciplinary study in which it is required the interaction of experts of several different fields of research including coupling of computer models with different standards and formats for input data and results. The complexity of a safety assessment, including the interaction between experts from several different fields of sciences, makes it very difficult to accommodate the uncertainties, that stem from several different sources, in a common framework. For example, data that support calculations for useful lifetime of concrete structures provide different levels of evidence than data that support selection of parameters values on simulations of radionuclides migration in the geosphere or prediction of redox front movement rates. Therefore, a methodology is needed that is capable of analyzing uncertainties for all data sets in the same framework, including propagation and quantification of degrees of support and evidence for results. In 2004 a technical co-operation project was established between the National Nuclear Energy Commission, Brazil, and the STUK- the Radiation and Nuclear Safety Authority, Finland, for the development of an alternative study of uncertainty through the application of fuzzy sets and evidence theory. The objective of this project is to study the various sources of uncertainties, objective and subjective, and their effects on decisions throughout the high level waste disposal safety assessment. In this work some results of the project are presented.

1. Introduction

Traceability and clarity of the arguments is very important for the acceptance of the safety assessment of a waste disposal facility. The safety assessment process should be able to provide stakeholders with a clear picture of the whole process of decision making, including degrees of confidence and conservatism, with which decision were taken and their effects on the final result.

According to [1] there is a requirement of reasonable assurance that the calculated doses will be within standard limits. This statement is a recognition of the impossibility of obtaining results with absolute certainty.

The complexity of the analysis, in addition to lack of data and ignorance, are major sources of subjective uncertainties in safety assessment. There is a need for a methodology that is capable of integrating traditional approaches for uncertainty analysis, such as probability, to innovative ones, such as fuzzy sets.

Evidence theory supposes the existence of a universe of discernment, $\Theta$, which is comprised of subsets with all possible combinations of answers to a question. Each of these subsets will be assigned with a basic probability and a degree of fuzziness. In order for this methodology to be used, all the available information, including those in form of linguistic expressions, have to be translated into a collection of propositions. This collection of propositions, which constitute the available knowledge, are then represented by mathematical expressions so they can lend themselves to calculations.
2. Knowledge representation

The expert system will be developed in three steps. The first step towards the development of a safety assessment expert system, where both subjective and objective uncertainties are integrated into a common framework, is the arrangement of all the information and data into a knowledge based system.

Knowledge is a collection of propositions [2]. Very often information is represented by linguistic expressions, and therefore, the mathematically represented propositions should be capable of keeping the same quality of the original information. Fuzzy sets are designed to deal with vague and ambiguous data and information, and together with evidence theory, can be an important tool for the development of a realistic knowledge based system.

The next step of this study, after the development of the knowledge based system, will be the development of a rule-based system. In this system the mathematically represented propositions are in the form of if-then rules.

The third step is the development of an inference machine that can be used for mathematical simulations of the repository system.

3. Practical example

In this section a practical application of the evidence theory and fuzzy sets is presented. This is about the evaluation of the useful life of the cooper canister the principal technical barrier in the disposal concept adopted by, e.g., Sweden and Finland. Spent fuel is expected to remain radioactive for hundreds of thousands of years and there is no practical means to guarantee that the disposal facility and canisters will remains protective for this long time frame. The available information, laboratory tests and calculations based on thermodynamics, as well as qualitative evidence from observation of natural systems, are used as support for decisions regarding the performance of the canisters for that period of time.

Two approaches for safety analysis are presented according to [3] and [4]. The first approach is a combined mass-balance/transport model that calculates the maximum depth of general corrosion (mm) over some time period (100 + years). The second approach relies on a steady-state mass transport-reaction model that calculates the service life in years. Following Dempster-Shafer, or evidence theory, the two approaches correspond to two independent sources of information.

The frame of discernment, Θ, is comprised of all the possible processes, represented as propositions, that can lead to corrosion of the Cu canisters.

For each approach [3], the corrosion assessment was divided into an initial period of aerobic (oxic) corrosion, followed by a longer period of corrosion under anaerobic (anoxic) conditions. During both periods, general corrosion and pitting were assumed to be possible. A total corrosion rate was calculated as a sum of general and pitting corrosion.

For the pitting process, two limits were studied, a lower and realistic rate based on a pitting factor of 2 (PF=2) for a lower limit (L), and an upper or higher (H) conservative limit with PF=5.

Therefore, two limits for total corrosion, based on PF=2 and PF=5, were calculated. The results of this analysis is presented in table I [4].

Table I: degrees of belief and plausibility for corrosion rate limits for each approach *

<table>
<thead>
<tr>
<th>Range limits</th>
<th>Low</th>
<th>Upper</th>
</tr>
</thead>
<tbody>
<tr>
<td>Approach 1</td>
<td>bel₁ = 0.815</td>
<td>bel₁ = 0.735</td>
</tr>
<tr>
<td>Approach 2</td>
<td>bel₂ = 0.74</td>
<td>bel₂ = 0.74</td>
</tr>
<tr>
<td>Combined</td>
<td>bel₁₂ = 0.74</td>
<td>bel₁₂ = 0.74</td>
</tr>
</tbody>
</table>

*Table according to [4].
Where bel stands for belief function and pl stands for plausibility function. These concepts are related to the degrees of support that a certain proposition is assigned according to available evidence. Bel$_{12}$ refers to what is believed to be the degree of truth for the calculated result and pl$_{12}$ is the plausible value for the truth in the result.

4. Conclusions and next developments

Radioactive waste disposal facilities safety assessment is a very complex task. There are several sources of subjective uncertainty that are difficult to quantify. Innovative approaches for uncertainty analysis, such as the one here presented can help public and decision makers have a comprehensive understanding of the whole process. As a further development of the methodology the studies will be focused on problems mainly related to geochemistry and conservatism, i.e., problems related to uncertainties of radionuclide solubilities and sorption behaviour. In this next phase, parameters determined by probability and deterministic approaches will be studied. Deterministic values are, in some cases, assumed to be very conservative [3]. However, the degrees of conservatism are not always clear and consequently its real effect on the safety of the repository may not be clear as well. These values can be assigned with degrees of support or belief and, therefore, be treated as fuzzy numbers. The probabilistic results are usually given as a set of curves which maybe confusing to decision makers. Each curve, however, can have different degrees of support and, therefore, constitute a fuzzy set of answers. After this process of fuzzification, both deterministic and probabilistic approaches can be integrated into the same framework. One of the advantages of this methodology is a more effective analysis of uncertainty propagation and sensibility analysis.

REFERENCES


Development of a decision support system for siting a LILW and disused sealed sources disposal facilities in Brazil

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\textbf{Abstract}. The Brazilian National Nuclear Energy Commission, CNEN, is entrusted with the responsibility of safe management of radioactive wastes in Brazil. It has achieved this through stringent regulations, periodic inspections and regular collection and storage of waste from fuel cycle. On 20 November 2001, the President of Brazil approved Law 10308 which addresses site selection, construction and licensing, operation and enforcement of radioactive waste interim storage and final disposal in Brazil. Article 37, within this Law, demands that “CNEN must begin studies for site selection, project, construction and licensing, to start operation within a period of time as short as technically possible, of a final disposal facility within national territory”. Currently, at CNEN research institutes there are interim storage facilities, which can warehouse approximately 500 m\textsuperscript{3} of low and intermediate level radioactive waste that stem from a large range of nuclear applications. It is estimated that the nuclear waste would grow at a rate of 50\% every year for the next ten years. With this growth, the current capacity for interim storage will get depleted very quickly. In addition, about 8,000 disused sealed sources, formerly used in industrial gauge instruments, in research and in radiotherapy, have been collected as radioactive wastes. As part of the CNEN efforts to deal with the radioactive waste an IAEA technical co-operation project was established. This project aims at the development of an expert system as a decision support tool for selection of candidate sites for siting a national disposal facility for low and intermediate level institutional waste and spent sealed sources in Brazil. Due to limited funds for sites characterization, this expert system needs to be capable of dealing with vague and incomplete information while keeping a high degree of confidence in the results.

1. \textbf{Introduction}

Safety assessment of radioactive waste disposal facilities comprises activities such as site selection and characterization, siting, repository design and post closure performance assessment. Each one of these activities is comprised of several others. This makes safety assessment a multi-disciplinary study, which requires analysis of an extensive data base. As a consequence uncertainties will arise due to complexity, ambiguity and lack of data and ignorance. The uncertainty that may exist at every stage will get propagated through the final results. Despite the latest developments on computing capabilities, analysis of uncertainty propagation is still a problem in complex analysis such as this one.

As a first step a large region will be selected for study according to technical and social-economic criteria. Literature data will be used for this phase of analysis. On a second stage some field data will be collected and complemented by data provided by the national geological survey for a more detailed analysis. Experts’ judgement will have a strong role on data interpretation which will be used on mathematical simulations aiming at sites ranking.

The subjective component in data interpretation can pose difficulties on uncertainty propagation analysis and, consequently, on a realistic evaluation of their impacts on degrees of confidence or support that will lead to decisions regarding the suitability of sites for disposal of the waste. This project has an additional complexity due to the fact that the selected site will have to be fitted for two repositories, near surface and borehole, with different safety requirements. Therefore, a strong mathematical basis as a support for decisions, and consequent confidence, is needed.

2. \textbf{Objective}

The objective of this project is to provide decision makers with a clear picture of the suitability of sites regarding the siting of the two disposal units. This information will be used for site selection for
further characterization and confirmation. To this end it is necessary to have enough confidence in the
information, or evidence, that support a ranking of previously selected sites.

The decision making process should be as clear and simple as possible in order to be easily understood
by the public and decision makers while keeping the same quality of information.

Due to the complexity of the analysis and lack of data the support for decision may not be
straightforward. Therefore, the quantification of imprecise and ambiguous data can make the system a
powerful tool to deal with incomplete data.

This decision support system will be particularly useful in this project because it is hoped to make it
possible a comprehensive analysis, taking into account requirements for both repositories.

Although a borehole is simpler than geological repositories, some of its safety requirements are similar
to those of high level waste repositories. Therefore, a site that is suitable for a near surface repository,
may not be the best choice for a borehole. For example, the depth of the borehole will depend on the
types of sources and other geological and geochemical factors.

All of these factors have to be balanced in order to generate a list of suitable sites with similar degrees
of safety.

3. Methodology

According to Zadeh [1] knowledge is a collection of propositions, and in order to constitute
knowledge, the propositions have to be understood. Also, in order to serve as a basis for uncertainty
propagation analysis as a support for the decision making process, the propositions have to be
mathematically represented.

To this end evidence theory and fuzzy sets will be used. The available data and information are
translated into fuzzy sets and fuzzy probabilities that together with belief and plausibility functions
will make it possible to get all types of information into a common framework.

Degrees of membership can represent the degrees of truth in the propositions. For example, a too
conservative proposition would mean the process being analyzed is not well understood and
consequently it would have a higher degree of fuzziness, or lower degree of membership to a belief
function.

This methodology can help on the establishment of a common quantification of degrees of truth or
belief in the evidence for the very different types of data from different sources, and therefore, a basis
for sites ranking is possible. A first version of this system is expected to be delivered by the end of
2005.

4. Short example of application

Fuzzy sets have flexible boundaries (i.e., an object has variable degrees of membership to the set, from
zero to one) as opposed to crisp sets, which have crisp or sharp boundaries (i.e., the degree of
membership of an object is either zero or one).

In safety assessment, complex natural processes are usually simulated by simplified models. One
example is Kd, distribution coefficient, which is a function of the type of soil (sand, clay, silt, organic,
etc), pH, state of oxidation of the radionuclide, among other factors.

This parameter has a strong degree of temporal and spatial variability. In order for the results to be
presented as valid probability distributions, a considerable amount of data would have been required.
Typically, either deterministic values, or probability distribution, are chosen with the help of experts’
judgement. In other words, this is a source of type B, or subjective, uncertainty.

Kd can be represented as a fuzzy number or fuzzy set and defined based on a fuzzy classification
process. These sets could be “High Kd”, “Medium Kd”, and “Low Kd”, for example. A membership
function will represent the degrees of truth, according to available data and experts’ opinion, that a
certain Kd value has to a specific set.
With this approach it is possible to represent the vagueness and ambiguity usually encountered in environmental data. Its main advantage is that it allows the experts to use natural language to express their interpretation of site conditions, and yet with a strong mathematical basis.

5. Conclusions

Brazil and IAEA have started a technical co-operation project for the development of a decision support system capable of analysis of vague and incomplete data in order to assist on the siting of a low and intermediate waste repository and a borehole.

Limited funds for data collection and complexity of analysis require an innovative method of data analysis and interpretation as decision support. The main challenge is to bring very different data formats, and consequent uncertainty propagation, into a common framework.

A methodology based on fuzzy sets and evidence theory will be used on the development of the decision support tool.

REFERENCE


BIBLIOGRAPHY


Results of safety reassessment of repositories for institutional radioactive waste
A Czech Republic Case

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Abstract. Respecting the change of the ownership of radioactive waste repositories in the Czech Republic, RAWRA had to apply for new licenses of operated repositories. This process had to be supported by updated safety assessments that should have improved the knowledge of disposal systems performance and become a base for update of operational conditions including acceptance conditions for the radioactive waste disposal. In the case of repositories Richard and Bratrství, which have been used for institutional waste disposal, safety re-assessment included a wider range of evaluated scenarios and provided a background to the specification of repositories closure.

1. Introduction
With the transfer of Czech repositories to the ownership of state, it was necessary to update their licences with respect to legislative changes. The process was more difficult in the case of repositories of institutional waste that have been operated for tenths of years in conditions reflecting the prior state controlled by former owner and previous state of legislation.

2. Safety re-assessment context
Repository Richard and Bratrství have been both located in near surface abandoned mines, limestone and uranium, respectively. Richard repository was designed to accept institutional waste containing artificial radionuclides, Bratrství repository has to contain natural radionuclides.

2.1. History
Acceptance of institutional waste to the repository has been controlled by individual dose limit 1 mSv/yr and by additional constraints, reflecting transport and manipulation needs. In past, Richard and Bratrství repositories were more times subjected to safety analysis, with an available extent of input data. The safety assessment method was relevant to the stage of knowledge and to regulatory requirements. With the change of legislation and progress in safety assessment methodology, it was necessary to improve the understanding of repository system performance and to evaluate more relevant scenarios supported by newly obtained input data.

2.2. Objectives
The objectives of reassessment of Richard and Bratrství repository were: update of inventory data, acquisition of data from the sites supported by hydro-geological survey, review of scenarios set based on FEPs list, verification of models and safety calculations using various tools for describing disposal system performance, comparison of results with legislation requirements. By the regulation on radiation protection, the current individual dose limit has been set to 250 $\mu$Sv/yr.

3. Safety reassessment procedure
3.1. Revision of inventory
Safety reassessment was carried out using inventory data reviewed from historical shipping documents and from estimated future production of waste. In the case of uncertainties, there were used conservative inventory values.
3.1.1. Richard repository

Since 1964 till 2004, in the repository 5100 m$^3$ of radioactive waste (more than 60% of the capacity) was disposed containing a total activity is $1 \times 10^{15}$ Bq (see Table 1). In majority, waste has been placed in double 200 l barrels and/or 60 l barrels. In there is present about 100 of radionuclides. Most important from the safety point of view are $^{239}$Pu, $^{241}$Am, $^{60}$Co, $^{137}$Cs, $^{90}$Sr, $^{14}$C, $^{3}$H.

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Activity in the repository in 2004 [Bq]</th>
<th>Activity limit [Bq]</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^3$H</td>
<td>$5.95 \times 10^{13}$</td>
<td>$1.10^{15}$</td>
</tr>
<tr>
<td>$^{14}$C</td>
<td>$7.22 \times 10^{12}$</td>
<td>$1.10^{14}$</td>
</tr>
<tr>
<td>$^{36}$Cl</td>
<td>$1.48 \times 10^{9}$</td>
<td>$1.10^{10}$</td>
</tr>
<tr>
<td>$^{90}$Sr</td>
<td>$3.37 \times 10^{12}$</td>
<td>$1.10^{14}$</td>
</tr>
<tr>
<td>$^{99}$Tc</td>
<td>$1.66 \times 10^{8}$</td>
<td>$1.10^{10}$</td>
</tr>
<tr>
<td>$^{129}$I</td>
<td>$9.10 \times 10^{3}$</td>
<td>$1.10^{8}$</td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>$4.20 \times 10^{14}$</td>
<td>$1.10^{15}$</td>
</tr>
<tr>
<td>long lived alpha</td>
<td>$1.40 \times 10^{13}$</td>
<td>$2.10^{13}$</td>
</tr>
</tbody>
</table>

Table 1. Inventory of Richard repository

3.1.2. Bratrstvi repository

Since 1974 till 2004, in the repository 900 m$^3$ of radioactive waste (more than 70 % of the capacity) was disposed containing a total activity of $1,2 \times 10^{12}$ Bq. The most of the waste has been placed in 200 litre barrels. There are mostly natural radionuclides disposed.

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{210}$Po</td>
<td>$1.10^{2}$</td>
<td></td>
<td>$2.10^{1}$</td>
</tr>
<tr>
<td>$^{228}$Th</td>
<td>$1.10^{4}$</td>
<td></td>
<td>$5.10^{1}$</td>
</tr>
<tr>
<td>$^{228}$Ra</td>
<td>$5.10^{6}$</td>
<td></td>
<td>$5.10^{12}$</td>
</tr>
<tr>
<td>$^{210}$Pb</td>
<td>$2.10^{7}$</td>
<td></td>
<td>$1.10^{12}$</td>
</tr>
<tr>
<td>$^{226}$Ra</td>
<td>$1.2 \times 10^{12}$</td>
<td>$9.7 \times 10^{11}$</td>
<td>$5.10^{12}$</td>
</tr>
<tr>
<td>$^{239}$Pu</td>
<td>$3.7 \times 10^{10}$</td>
<td></td>
<td>$5.10^{11}$</td>
</tr>
<tr>
<td>$^{234}$U</td>
<td>$3.4 \times 10^{11}$</td>
<td>$3.3 \times 10^{11}$</td>
<td>$2.10^{12}$</td>
</tr>
<tr>
<td>$^{235}$U</td>
<td></td>
<td></td>
<td>$1.10^{12}$</td>
</tr>
<tr>
<td>$^{238}$U</td>
<td></td>
<td></td>
<td>$1.10^{12}$</td>
</tr>
<tr>
<td>$^{212}$Th</td>
<td>$1.10^{3}$</td>
<td>$2.3 \times 10^{8}$</td>
<td>$3.10^{12}$</td>
</tr>
<tr>
<td>total alpha</td>
<td></td>
<td>$1.3 \times 10^{12}$</td>
<td>$1.10^{13}$</td>
</tr>
</tbody>
</table>

(*) values used in the safety assessment

Table 2. Inventory of Richard repository

3.2. Model development and calculation tools

3.2.1. Richard repository model

The choice of scenarios has been based on test calculations of the source term describing possible ways of release of radionuclides to groundwater affected by waste form of the waste and by possible filling options. Far field calculations were specified by newly obtained field date form bore holes and field tests. FEPs methodology was used to develop...
normal, alternative and critical scenarios,
update of hydro-geological model of the site,
biosphere model,
intrusion scenario.

The sensitivity analysis has been principally built on the source term construction, depending on inventory composition, final waste form characterisation and repository closure options, including assessing degree of saturation. There were assessed more than ten source term variants, with two principal pathways in geosphere and two possibilities of future use of the site. Calculation and model construction uncertainties were minimized using parallel calculations by various codes (MASCOT, GOLDSIM, SUTRA). The results of calculations of effective doses are brought out in Table 3.

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Normal evolution</th>
<th>Intrusion</th>
<th>Possible limit</th>
</tr>
</thead>
<tbody>
<tr>
<td>Source term</td>
<td>Effective dose</td>
<td>Limit of activity in a 200 l drum</td>
<td>Limit of total activity</td>
</tr>
<tr>
<td>[Bq/yr]</td>
<td>[Sv/yr]</td>
<td>[Bq]</td>
<td>[Sv/yr]</td>
</tr>
<tr>
<td>^3H</td>
<td>2.0\times10^-10</td>
<td>4.21.10^19</td>
<td>3.24.10^-19</td>
</tr>
<tr>
<td>^14C</td>
<td>1.0\times10^-9</td>
<td>2.04.10^-19</td>
<td>1.02.10^-9</td>
</tr>
<tr>
<td>^89Sr</td>
<td>8.0.10^-14</td>
<td>8.62.10^-24</td>
<td>5.89.10^-14</td>
</tr>
<tr>
<td>^137Cs</td>
<td>1.0.10^-33</td>
<td>1.35.10^-86</td>
<td>1.42.10^-33</td>
</tr>
<tr>
<td>^239Pu</td>
<td>2.0.10^-6</td>
<td>2.02.10^-14</td>
<td>4.03.10^-6</td>
</tr>
<tr>
<td>^226Ra</td>
<td>3.0.10^-6</td>
<td>1.18.10^-17</td>
<td>2.10.10^-6</td>
</tr>
</tbody>
</table>

Table 3. Effective doses calculated for derivation of activity limits in Richard repository.

The actual limits of operation and acceptance condition have been set by the State Office for Nuclear Safety more restrictively, that would be necessary considering the results of safety assessment, respecting possible waste forms and transport conditions.

3.2.2. Bratrstvi repository

Because of special historical and geological conditions of the repository, safety assessment has been based on the source term scenarios construction. Far field pathway is missing because of proximity of the repository space and the mine water system. Source term calculations have been carried out for principal closure options (filling/no filling) and two possible sites of water withdrawal in the environment. In situ measurement of water infiltration, host rock geo-technical stability and drainage effectiveness are carried out continuously to support near field modelling. Because of the uncertainty in inventory solubility, the normal evolution scenario has been constructed for various post closure time frames respecting more probable ways of host rock and filling degradation. As a result, the effective dose is two orders higher in the intrusion scenario than in transport scenarios, i.e. normal evolution, alternative and emergency scenarios. The maximum dose from the intrusion scenario is 1.8.10^-4 Sv as is shown in Table 4.

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Source term</th>
<th>Effective dose</th>
<th>Limit of activity in a 200 l drum</th>
<th>Limit of total activity</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>[Bq/yr]</td>
<td>[Sv/yr]</td>
<td>[Bq]</td>
<td>[Bq]</td>
</tr>
<tr>
<td>^226Ra</td>
<td>6.2E+8</td>
<td>2.3E-5</td>
<td>1.10^-9</td>
<td>5.10^-4</td>
</tr>
<tr>
<td>^210Pb</td>
<td>6.2E+8</td>
<td>5.6E-5</td>
<td></td>
<td></td>
</tr>
<tr>
<td>^210Po</td>
<td>6.2E+8</td>
<td>9.7E-5</td>
<td></td>
<td></td>
</tr>
<tr>
<td>U</td>
<td>8.0E-8</td>
<td>1.10^-7</td>
<td>2.10^-4</td>
<td></td>
</tr>
<tr>
<td>^232Th</td>
<td>3.6E+5</td>
<td>1.2E-8</td>
<td></td>
<td></td>
</tr>
<tr>
<td>long lived alpha</td>
<td></td>
<td></td>
<td>1.10^-7</td>
<td>1.10^-13</td>
</tr>
</tbody>
</table>

Table 4. Effective dose from intrusion and current acceptance conditions.
4. Licensing issues and conclusions

No need has arisen to restrict acceptance conditions with regard to the formerly disposed waste. In the case of Richard repository, the list of limited radionuclides was upgraded and there were included long lived beta gamma radionuclides, not limited before. In Bratrstvi repository, it is strongly needed to dispose only natural radionuclides, with respect to the character of the site and to formally check the uncertainties in the inventory description.

Current safety assessment results were laid down in closure options proposal. The preference is declared to follow the way of optimum filling of both the repositories in the process of their closure. Repository safety could be improved by at least one order. Actually, there are in process more activities assessing potential filling materials and finding special technologies of their application, explicitly a hydraulic cage application in Richard repository and concrete filling combined with bentonite or clay barrier in Bratrstvi repository.

REFERENCES

Uranium migration in sand soil – controlled experiments

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Israel

Abstract. Study of Uranium migration in porous media such as sandy soil is essential to evaluate the potential contamination of groundwater situated in the vicinity of radionuclide waste disposal sites and nuclear facilities. At first, the infiltration of uranium solution through local sandy soil focuses on the influence of solution pH and the grain size ranges of the sandy soil on adsorption and leaching values. Solutions with three pH levels of 0.5, 3.6 and 9 were allowed to infiltrate through columns filled with sand of three grain size fractions: 63-125 µm, 125-250 µm and 250-500 µm. The pH level was found to be a more important variable than grain size. In alkaline solution colloids were produced, which are more mobile between the pores, so uranium was found at deeper layer. The second experiment involved leaching of uranium from the solid material. After the equivalent of 7,000 mm of rainfall had percolated through the column, only 1/3 of the uranium had been leached. This result indicates that only negligible leaching occurs in arid and semi-arid zones.

1. Introduction
Study of uranium ion migration in porous media such as sandy soil is essential in the evaluation of potential contamination of the groundwater present in the vicinity of radionuclide waste disposal sites and nuclear facilities. Studies of uranium ions' and other radionuclides' migration were carried out mostly in areas polluted by nuclear activities, such as Hiroshima, Chernobyl, NTS (Nevada Test Site) and Kazakhstan. These studies obviously dealt with 239Pu, 235U, 137Cs and 90Sr. In Ref. [2] found in the NTS region, an increase in 235U concentration on the surface which decreased with depth, while no change in 238U concentration was found on the surface or with depth. Their explanation was that while 235U resulted from nuclear tests, 238U comprises natural uranium which is an ingredient in the local soil. The same trend was found after the criticality accident in Tokai-Mura, Japan [Ref. 3]. Concentrations of 238U and 235U, resulting from the accident, decrease with depth, while the thorium isotope concentration indigent to the area remained unchanged. New research [Ref. 1] deals with depleted uranium from military ordnance on arid soil through a series of batch sorption experiments. The authors reported that subsurface sorption is inversely proportional to soil pH and directly proportional to soil clay content and cation exchange capacity.

2. Theoretical considerations
The decay chain of 238U to radon is shown below.

\[
\begin{align*}
238U & \rightarrow 234Th \rightarrow 234Pa \rightarrow 234U \\
\alpha & \quad 4.5 \times 10^9y & \beta^- & \quad 24.1y & \beta^- & \quad 1.7m & \alpha & \quad 2.5 \times 10^5y \\
230Th & \rightarrow 226Ra \rightarrow 222Rn \\
\alpha & \quad 8 \times 10^4y & \alpha & \quad 1600y & \alpha & \quad 3.8d
\end{align*}
\]

For uranium ore this chain is in equilibrium, while for uranium waste the radioactivity increases with time through the daughters. The main hazard from the presence of uranium in soil is emanation of radon gas. Three factors control radon seepage in porous media:

1. the half life of 3.8 days;
2. porosity of the sand, which controls the diffusion rate.
3. depth of the uranium waste disposal.
The depth of uranium is the only controlled parameter. The actual depth of the waste is influenced by surface erosion and leaching rate. This raises the following questions:

1. How far does the uranium migrate before being adsorbed by the soil?
2. What is the leaching rate?
3. What is the surface erosion rate of the cap and the sand material above the waste?

These parameters determine how deep waste should be buried. This research focused on understanding the adsorption and leaching rate.

Migration of uranium ions, like any other cations in the sub-soil, depends on many parameters of the solution as well as of the solid media through which they migrate. Some of the major factors are:

- The solution chemistry: pH and concentration of other ions.
- Soil composition: mineralogy, pH, amount of: clay, limestone and organic material.
- Soil physical properties: grain size, density and porosity.
- Soil water content: saturated or unsaturated.
- Amount of rainfall versus evapo-transpiration rate.

Uranium ions exchange between the solution that contains them and the sand through which penetrates being the main process retarding the migration of the ions in wet soil, relative to the advance of pure water.

In acid solution the uranium is in the form of UO₂(NO₃)₂(aq). In that form the uranium is completely dissolved. In alkaline solution and in the presence of sodium, uranium forms the insoluble salt Na₂U₂O₇:

\[
2\text{UO}_2(\text{NO}_3)_2(\text{aq}) + 6\text{NaOH} \rightarrow \text{Na}_2\text{U}_2\text{O}_7(\text{s}) + 4\text{NaNO}_3 + 3\text{H}_2\text{O}
\]

In low concentrations of uranium, the crystals of this salt are very small and the solution becomes a slurry. In that case the soil filters the uranium crystals out of the mobile liquid.

The soil is alkaline because of the presence of carbonates. Even with initial pH of 3.6, Na₂U₂O₇ precipitates in the soil. When the pH is higher than 9, carbonate anions are liberated from the soil and form a soluble complex with the uranium:

\[
\text{Na}_2\text{U}_2\text{O}_7(\text{s}) + 6\text{CO}_3^{2-} + 3\text{H}_2\text{O} \leftrightarrow 2[\text{UO}_2(\text{CO}_3)_3]^{4-}(\text{aq}) + 2\text{Na}^+ + 6\text{OH}^-
\]

But, simultaneously, alkalinity shifts the equilibrium in the opposite direction, hence limiting the concentration of the uranyl carbonate.

3. Experiments

The experiment began with measurement of the adsorption of uranium from a solution into the ground, followed by leaching the adsorbed uranium by artificial rain water (ARW). The aim of this experiment was to simulate penetration of uranium waste into the ground and its later subsurface migration by water. The influence of two important parameters, the pH of the solution and the grain size of the sand soil, on processes of adsorption and leaching were studied in a controlled experiment carried out in the laboratory.

The concentration of uranium in the sandy soil was measured by counting the emission of 186 keV gamma photons following the decay of ²³⁵U.

3.1. Adsorption

Soil samples that were taken from Yamin Plateau, contain over 90% quartz mineral, about 0.1% organic matter (reflection of the arid conditions in the area), and the rest is mostly CaCO₃. The sand was sifted in accordance with three grain sizes of 250-500 μm, 125-250 μm and 63-125 μm were selected for the measurements. Artificial rain water (ARW) was used to prepare the solution. The sand was introduced into a plastic column 3 cm in diameter and 120 cm long filled to 54 cm. 250 ml of UO₂(NO₃)₂ solution in three pH levels, containing 75 mgU/l of natural uranium were pored slowly and evenly into the column. 250 ml are equivalent to a rainfall of 35 mm. The drained water was collected
from the bottom of the column for pH analyses and gamma ray counting. The sand in the column was divided into 6 samples of 9 cm each which were analyzed individually by gamma ray counting.

3.2. Leaching

In order to imitate realistic infiltration of uranium waste into the ground, a solution of 250 ml containing 20 mg of UO$_2$(NO$_3$)$_2$ with pH= 3.6 slowly infiltrated into the sand column composed of typical local sand. As was found previously for this pH, all the uranium was adsorbed in the sand column. 50 litters of ARW, which represented 7,000 mm of rain, infiltrated during 72 days through the sand column with constant hydraulic head, to achieve steady state. The water was analyzed for uranium concentration after filtering through the column. The next step comprised the analyses of the sandy soil by gamma ray counting to obtain the final uranium distribution with depth and to ascertain the mass balance.

4. Results and discussion

4.1. Adsorption

The variation with depth of uranium concentration (uranium quantity relative to soil’s mass) for each of the three pH levels and three grain size ranges is shown in Table 1. The statistical error is on the order of 15%. The liquid phase drained from the column was collected periodically and its pH and gamma intensities were measured.

4.1.1. Uranium mass balance

Calculation of the mass balance for each experiment was undertaken to verify that the gamma ray counting and chemical analyses in the sands and in the drained water which left the sand column match the uranium input. In all the experiments the mass balance between the input and the output of Uranium fit within 15%, which will be considered as the experimental error.

Table 1: Changes in Uranium distribution in soil samples with different pH, grain sizes and depth (% of Uranium).

<table>
<thead>
<tr>
<th>Depth (cm)</th>
<th>1st experiment pH=0.6</th>
<th>2nd experiment pH=3.6</th>
<th>3rd experiment pH=9.0</th>
</tr>
</thead>
<tbody>
<tr>
<td>9</td>
<td>1.6% 3.7% 9.8%</td>
<td>96.3% 75.5% 44.0%</td>
<td>50.6% 67.2% 95.9%</td>
</tr>
<tr>
<td>18</td>
<td>0.5% 5.1% 9.5%</td>
<td>1.7% 15.3% 23.3%</td>
<td>43.0% 17.8% 2.6%</td>
</tr>
<tr>
<td>27</td>
<td>1.4% 5.2% 79.5%</td>
<td>0.6% 2.5% 20.0%</td>
<td>4.5% 11.6% 0.8%</td>
</tr>
<tr>
<td>36</td>
<td>0.8% 5.8% 0.5%</td>
<td>1.3% 2.3% 12.6%</td>
<td>1.9% 2.4% 0.0%</td>
</tr>
<tr>
<td>45</td>
<td>1.7% 5.7% 0.3%</td>
<td>0.1% 2.3% 0.0%</td>
<td>0.0% 0.4% 0.6%</td>
</tr>
<tr>
<td>54</td>
<td>2.8% 20.8% 0.3%</td>
<td>0.0% 2.0% 0.0%</td>
<td>0.0% 0.6% 0.0%</td>
</tr>
<tr>
<td>out</td>
<td>91.2% 53.8% 0.1%</td>
<td>0.0% 0.0% 0.0%</td>
<td>0.0% 0.6% 0.0%</td>
</tr>
</tbody>
</table>

4.1.2. Uranium distribution with depth

Analysis of Table 1 indicates as expected that the uranium concentration decreases exponentially with depth. The pH value of the solution turned out to be a very important parameter. When the acidity level of the solution is high enough to overcome the alkalinity of the soil, most of the uranium will not adsorb in the soil. When grain size range between 250-500 µm in pH level of 0.6, 2/3 of the Uranium migrated in the water, while in pH levels of 3.6 and 9, all the uranium was adsorbed along the sand column.
In alkaline solution and in the presence of sodium, uranium forms the insoluble salt Na$_2$U$_2$O$_7$. When the grain size is small enough the crystals will not penetrate the soil. When the grain size is bigger than the small crystals of Na$_2$U$_2$O$_7$ penetrated into the soil and reacted with the carbonate from the soil. The uranium then dissolved in the water phase as an anion complex ([UO$_2$(CO$_3$)$_3$]$^{4-}$ (aq) ) and as colloids and moved on to deeper layers.

### 4.2. Leaching

Calculation of mass balance was undertaken to verify the consistency of gamma ray counting and chemical analyses. The drained water contained 6.2 mg uranium, while in the sand 11.07 mg remained giving a total output of 17.27 mg, while the input was 20 mg. Hence the experimental error amounts to 15%.

Variations of uranium concentration in the drained water are shown in Fig. 1. This figure demonstrates the characteristics of the leaching process. During the first 15 days, no uranium was found in the drain water. This observation clearly proves the retardation effect experimentally. From the 16th day, the uranium concentration increased steadily up to the 30th day. From that point it decreased until the measurement was discontinued. A possible explanation for this phenomenon is that the adsorption strength between the uranium and sandy soil is inversely proportional to uranium concentration in the solid. So the higher the concentration, the faster the leaching process. This finding indicates that once uranium is adsorbed to a solid phase, water cannot leach it entirely. This may also explain the presence of uranium in top soils across the globe.

Figure 2 compares the original adsorption of uranium from a solution into clean soil, with the leaching of uranium from this soil. In this figure the uranium presence in the solid is measured as a function of depth. The figure clearly shows that leaching pushes the uranium deeper. This experiment represented leaching in a humid climate with abundant rainfall. Obviously, for arid and semi-arid climates, leaching would be significantly reduced and even negligible.

![FIG. 1: Variation of uranium concentration in the drained water as a function of volume and time.](image1)

![FIG. 2: Distribution of Uranium concentration with depth before and after leaching.](image2)
5. Conclusions

Based on this work, which measured the consequences of changing pH levels and sand grain size, pH level was found to be a very important variable. When the acidity level of the solution is high enough to overcome the alkalinity of the soil, most of the uranium will not adsorb in the soil. In weak acid pH level of 3.6, all the Uranium was adsorbed mainly in the top layer of the sand column. In alkaline pH level of 9 and large grain size, colloids play a major role changing the uranium distribution with depth. Small grain size means large surface area and better chemical interaction with the solution, but it also means better filtration ability of the precipitated crystals. This dual role of the grain size is expressed well in the results.

The experiment simulated humid climate (7000 mm of rain during 72 days), and yet only 1/3 of the uranium was leached within 54 cm. In arid and semi-arid climates leaching of uranium from the soil should be negligible.

The leaching process pushes some of the uranium deeper into the ground, but can never leach it completely.

REFERENCES


Using biological soil crust (BSC) for soil surface stabilization

Field experiments

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Abstract. The paper addresses the use of biological soil crust (BSC) as a top cover for radioactive waste disposal site for reducing dispersion of contaminants to the environment. The crusts reduce surface erosion and water percolation. Biological soil crusts are friendly to the environment and have longevity and durability in comparison to other geo-textile layers. The authors signed this idea as Patent no. 168527.

1. Introduction

The present research relates to an environmental friendly method for enhancing the formation or recovery or microfloral soil crust, particularly in human disturbed area or contaminated area (signed as patent no. 168527). The BSC (= biogenic, microphytic) reduces water percolation and soil erosion and therefore can be used to isolate waste disposal site from the environment. The main idea of using local and environmental materials that are friendly to the environment and therefore has longevity and durability. The authors suggest replacing geo-textile layer with the biological soil crust. The main advantage is the longevity of soil crust in compare to geo-technique layer. Soil crust will survive even in case of climate change or colonizing different habitats from humid to hyper arid zones around the world.

2. The creation of the crust

The biogenic soil crust, being usually between 1 and 15 mm thick [1], and often covering up to 70% of the arid or semi-arid area of the world is inhabited predominantly by cyanobacteria, soil bacteria, algae, and lichens, which may be accompanied by mosses in more humid areas, the species composition varying from place to place [2,3]. Algae and cyanobacteria produce mucilaginous polysaccharides that seal the surface and form the crust [4].

The most important among the crust organisms are cyanobacteria, called also blue-green algae, organisms and primary colonizer within the crust succession that have been present on the Earth for at least 3.5 billion of years, and have contributed the Earth's oxygen atmosphere. Cyanobacteria colonize the soil surface and may move during dry period under the surface. The soil particles are glued together by polysaccharides secreted by cyanobacteria and green algae, and form a hard layer [4]. Filamentous cyanobacteria, such as Microcoleus sp., are especially efficient in the crust formation in arid environments [2]. The soil particles are immobilized by being both glued to each other and entrapped in the web of filaments, wherein the cyanobacterial basis may be further strengthened by green algae, lichen species, or other plants, according to the precipitation amount of region [3,1]. From among lichens, which are algae or cyanobacteria symbiotically living with fungi, Colema sp. may be named as a typical representative, important in microfloral crusts. Mosses are contributing in more humid region with their rhizoids that been used as skeleton for fixing the crust to the soil [2], allowing increasing the crust thickness.

Biological soil crusts are contributing soil resistance to wind and water erosion. These crusts are important source of fixed carbon as they are the primary producer [6] and by to enrich the soil by nitrogen fixation [3,5].
3. Objective
To provide a new environment friendly method for isolating a waste disposal site by accelerating the colonization of native micro-organisms to the treated environment, for preventing contamination by surface erosion and water percolation.

4. Methodology
The present invention provides a method for enhancing the soil crust formation, comprising steps of:
   i) collecting runoff water from the surface of the soil (source soil) on a first site (source site), which source soil is covered with a microfloral crust, thereby obtaining a suspension that contains microfloral propagules enabling the germination of the crust microflora;
   ii) optionally modifying the concentration of propagules in suspension from the previous step; and
   iii) concentrating the suspension by evaporation or producing propagule powder by drying or freeze-drying for easy movement from site A to site B.
   iv) applying the propagules onto the soil (target soil) at a second site (target site), which target soil lacks a microfloral crust or is covered by a crust less developed than the crust on source soil, wherein source site and target site are in the same geographical and climate region; thereby accelerating the colonization of target soil by micro-organisms native to the environment. A biogenic crust can be easily assessed, both from the viewpoint of the layer thickness, and richness of organisms, and it is a simple routine techniques for applications, so that it will be easy to compare two microfloral crusts.

Several methods were used in the literature to measure succession within the biological soil crusts (crust cover [2], reflectance [7,8] morphological groups [9], polysaccharides and chlorophyl production [10,11].

5. Description of field experiments
In order to check preferred ways of utilizing the new method, the authors examined how the organisms, useful for soil stabilization, were spreading under the conditions of the Northern Negev via water and aeolian accumulation. Rainwater runoff and dust were checked for the content of particles involved in the reproduction of the relevant crust organisms. For this purpose, 16 stations were established for collecting dust, and 30 experimental plots were established in two-difference habitats. In each one, 5 plots initially having the crust and 10 initially lacking the crust, where 5 of them were used for reference. In the experiments, the study, focus on the way to enhance or recover the microfloral soil crust.

6. Preliminary results
The initial laboratory test (Fig. 1) shows how the soil crust prevented the water to infiltrate through the soil crust (Fig. 2). From the field cells results we already gain runoff in compare to zero in the uncrusted cells.

Another method was used to monitor successional trend is by CO₂ assimilation in the light of crust samples from a Negev-dune, under controlled laboratory conditions [6]. It was shown (Fig. 3), that the CO₂ assimilation was clearly affected by the successional stage of the biological crusts (e.g. 1. well developed, 2. semi-developed and 3. sub-semi-developed BSC, and 4. moving sand dunes.).

These preliminary results suggest that using biological soil crust, as a top cover to near surface radioactive waste disposal facility that proposed in this paper is very promising.
FIG. 1: Schematic illustration showing the infiltration test.

FIG. 2: Water percolation with and without biological soil crust.

FIG. 3: CO₂ assimilation in the light of crust samples from a Negev-dune, under controlled laboratory conditions. Light = 500 µmol photons m⁻² s⁻¹ (PAR); T = 22°C. Data shown are means ± SD of 3 replicates of 5 min measurements of a crust sample.
7. Future applications

Use of BSC as a top cover for a radioactive waste disposal site for reducing dispersion of contaminants to the environment, and for reducing surface erosion and water percolation, is friendly to the environment and has longevity and durability in comparison to other geo-textile layers.

REFERENCES


Regulatory framework for the radioactive waste disposal in Romania

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Abstract. A regulatory framework for the radioactive waste disposal in Romania is under development. The paper contains a short description of the regulations on the radioactive waste management as well as a description of the new regulations which will be issued by the regulatory authority.

1. Introduction
In Romania the radioactive waste results from different activities such as: production of nuclear energy, use of radioisotopes in medicine, agriculture, industry and from research. Also, the extraction and preparation of nuclear fuel from uranium ore represents a main source of radioactive waste.

According to the Law no. 111/1996 on the safe deployment of nuclear activities, with the subsequent completions and modifications, republished in 2004, the National Commission for Nuclear Activity Control (CNCAN) is the competent authority in the nuclear field.

Under the above Law, CNCAN is the regulatory and authorization body, having responsibilities in the field of nuclear safety, radiation protection, radioactive waste management, quality assurance, physical protection, emergency preparedness, safeguards, export control, transport, operator certification, nuclear liability, international co-operation with foreign counterparts and expert international organisations, and strict observance and enforcement of Romania’s international commitment to the peaceful uses of nuclear energy under bilateral co-operation agreements and contracts, etc. This Law also establishes the competencies, responsibilities and obligations of CNCAN and other organisations and institutions involved in the nuclear field. This Law applies to all nuclear activities, including the activities relating to radioactive waste management.

In order to manage the nuclear activities in safe manner the CNCAN as regulatory authority is empowered to issue regulations for detailing the safety requirements.

2. Laws on radioactive waste management
2.1. Law No. 320/2003
Law No. 320/2003 for approval of Government Ordinance no. 11/2002 on the management of spent nuclear fuel and radioactive waste, including their final disposal, was published in the Official Law Bulletin of Romania.

The Romanian Government has issued the Government Ordinance no. 11/2003 on the management of spent nuclear fuel and radioactive waste, including their final disposal, in order to fulfil its obligations according to Law 105/1999.

The law 320/2003 establishes the National Agency for Radioactive Waste (ANDRAD) as waste management authority for co-ordination of activities related to radioactive waste.

The main duties of ANDRAD are the following:

- co-ordinates the activities of management of nuclear spent fuel and radioactive waste;
- is responsible for disposal of nuclear spent fuel and radioactive waste;
- Issues the national strategy for safe management of nuclear spent fuel and radioactive waste;
- creates and maintains the national data base regarding the spent fuel and radioactive waste;
- analyzes the characteristics of spent fuel and radioactive waste in view of their management.
The law is applied to the safe management of the nuclear waste resulting from both nuclear fuel cycle and from non nuclear activities.

Based on the duties of ANDRAD the Nuclear Agency issued the Ordinance 844/2004 on the national strategy regarding the management of spent nuclear fuel and radioactive waste on medium and long term. The strategy established the activities, responsibilities and time frame as well as an action plan for implementing of the national strategy.

2.2. Law No. 43/1995


By ratification of this Convention Romania became a contracting part and it must fulfil its obligations regarding the enforcement of the fundamental safety principles to the nuclear facilities, including the ones used for repository activities.

Achieving a world-wide high level of nuclear safety, by increasing the national security measures and the international co-operation in this area, represents the main objective of this Convention.

2.3. Law No. 105/1999


By ratification of this Convention Romania is assuming fully responsibility for the safe management of the radioactive waste, meaning the protection of occupationally exposed personnel, the public, environment and property, now and in the future.

The Convention is applied to radioactive waste safe management arising from civil applications and radioactive waste effluent discharges.

3. Regulations on the radioactive waste management

3.1. Radiological safety fundamental norms (NSR-01)

This regulation was approved by president of CNCAN by Order no. 14/2000 and published in the Official Law Bulletin of Romania.

This regulation is based on the Council Directive 93/26/EURATOM and on the IAEA Basic Safety Standards no. 115 on International Basic Safety Standards for protection against Ionizing Radiation and on the Safety of Radiation Sources. The regulation contains general radiological safety requirements. Also, the regulation details the concept of exclusion, exception and clearance and establishes the requirements, criteria and levels of exclusion and exception.

3.2. The fundamental norms on safe management of radioactive waste (NDR-01)

This regulation was approved by president of CNCAN Order no. 56/2004 and published in the Official Law Bulletin of Romania. The regulation is based on the IAEA SS no. 111-F “The principles of radioactive waste management”. This contains the principles of the safe management of radioactive waste as well as the requirements for fulfilling of these principles.

The regulation is applied to all activities involving safe radioactive waste management, from their generation to final disposal. The stages of radioactive waste management - pre-treatment, treatment, conditioning, interim storage (pre-disposal) and final disposal, are defined in an Appendix of these Norms.

3.3. Norms on clearance levels of radioactive materials arising from authorised nuclear activities (NDR-02)

The regulation was approved by the president CNCAN Order no. 62/2004 and published in the Official Law Bulletin of Romania. The regulation completes the requirements of Radiological Safety Fundamental Norms (NSR-01) and contains the methodology for approving by CNCAN of the
conditional or unconditional clearance of materials. The regulation does not contain the levels for
clearance. The levels for clearance are established by Radiological Safety Fundamental Norms (NSR-
01).

3.4. **Radiological safety norms for radioactive waste originated from uranium and thorium mining
and milling (NMR-02)**

These Norms were approved by president CNCAN Order no. 192/2002 and published in the Official
Law Bulletin of Romania.

The regulation is based on the IAEA safety standards and it completes the requirements of
Radiological Safety Fundamental Norms (NSR-01). This regulation contains the requirements for
siting, construction, operating and decommissioning of radioactive waste facilities from uranium
mining and milling.

3.5. **The set of norms for the safe transport of radioactive materials**

The set of the transport regulations contain four norms. The Fundamentals Norms for the safe transport
of radioactive materials (NTR-01) were approved by CNCAN Order no. 373/2001 and published in
the Official Law Bulletin of Romania. The norms are based on the IAEA TS-R-1 edition 1996 on the
requirements on the safe transport of radioactive materials.

The authorization procedure is detailed in the Norms for Transport of Radioactive Materials-
Authorization Procedure (NTR-04).

The Norms for the international shipment of radioactive materials involving the Romanian territory
(NTR-02), based on the Council Regulation 1493/93 on the shipment of radioactive substances
between Member States, contain the requirements for performing of shipments of radioactive materials
from/to European Union countries. The norms are fully in compliance with the council directive,
although Romania is not a Member State.

The Norms for international shipment of radioactive waste with involving the Romanian territory
(NTR-03) based on the Council Directive 92/3/EURATOM on supervision and control of shipment of
radioactive waste between Member States and into and out of Community contain requirements for the
international shipment of radioactive waste taken into consideration that the import of radioactive
waste on the Romanian territory is prohibited. For this reason the norms are not fully in compliance
with the directive mentioned above.

4. **Regulation on radioactive waste in plan to be issued by National Commission for Nuclear
Activities Control (CNCAN)**

According to the action plan for the implementation of the national strategy, Romania's current legal
framework (concerning radioactive waste management) needs to be completed.

In this way, CNCAN, as the national regulatory authority empowered to issue regulations on the
radioactive waste management, elaborated the following norms, currently being in the approval stage:

- Norms for the classification of radioactive waste;
- Norms on the general requirement for the near surface disposal of radioactive waste
- Norms on the siting of near surface disposal activities;
- Norms for pre-disposal management of radioactive waste;
- Norms on the release of radioactive effluents into environment.

4.1. **Norms for classification of radioactive waste**

According to the norms the classification of radioactive waste is performed by the generator. The
objective of these norms is to establish the requirements on radioactive waste classification. The
Norms propose a classification based on the IAEA SS 111-G-1.1 “The classification of radioactive
waste”. In this way, the radioactive waste shall be classified such: exempt waste, transition waste, very
low level waste, short lived low and intermediate level waste (LILW – SL), long lived low and
intermediate level waste (LILW – LL), high level waste (HLW). The activity limits for each
radioactive waste class are not specified within this regulation, going to be subsequently established by CNCAN.

4.2. **Norms for near surface disposal of radioactive waste**

This regulation is based on the IAEA SS WS-R-1 “Near Surface Disposal of radioactive waste”. This regulation shall contain the general requirements for near surface disposal of radioactive waste regarding the safety assessment, waste acceptance, characteristics of the site, design of disposal facilities, construction, operation and closure of the facility and on the post closure phase.

4.3. **Norms on the siting of near-surface disposal facilities**

This regulation is based on IAEA SS 111-G-3.1 “Siting of near surface disposal facilities”. The regulation establishes:

- General requirements regarding the siting of near surface disposal facilities
- Specific requirements for the four stages of the systematic siting process for a near surface disposal facility: conceptual and planning, area survey, site characterisation, site confirmation.

4.4. **Norms for pre-disposal management of radioactive waste**

This regulation is based on the IAEA SS WS-G-2.5 “The pre-disposal management of low and intermediate level radioactive waste”. The regulation establishes:

- general requirements for the radioactive waste management, during pre-disposal activities;
- requirements for the different stages of the radioactive waste management to be predisposed: generation, processing (pre-treatment, treatment, conditioning), pre-disposal (storage) and transportation.

The regulation shall apply to all the stages of radioactive waste management activities prior to their disposal and it shall not apply to the discharge of radionuclides to the environment, nor to disposal of radioactive waste.

4.5. **Norms on radioactive effluents releases into the environment**

This regulation shall complete the requirements of the Radiological Safety Fundamental Norms (NSR-01) and the Fundamental Norms on safe management of radioactive waste (NDR-01). The regulation shall set forth the principles and the general requirements on liquid and gaseous radioactive effluents releases in the environment, as well as the authorisation and monitoring (control) activities for these discharges. The regulation shall apply to all the facilities that, during normal operation, will generate environmental releases of liquid or gaseous radioactive substances, within limited quantities and concentration. The regulation shall not apply to the radioactive effluent releases arising from radioactive waste final repositories.
Regional activity on safe management of radioactive waste under the framework of the forum for nuclear co-operation in Asia (FNCA)

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Abstract. Nine Asian countries including Australia are participating in various projects among 8 fields under the framework of FNCA. Radioactive Waste Management (RWM) project is one of these projects. Annual workshop provides to member states a good and informative platform for enhancing improvement of good and safe management of radioactive waste. Through the past activities, this project published three reports, namely “RWM Consolidated Report”, “SRSM Activity Report”, and “Activity Report of TENORM Task Group”. RWM Newsletters are published twice a year, and provide a good information and communication among member states, too. Further, an internet communication through FNCA web-site is a good tool for mutual understanding and sends messages not only to FNCA RWM colleagues, but also to other international organizations.

1. General introduction of FNCA

The Forum for Nuclear Co-operation in Asia (FNCA) was established in 2000, after a future work discussion at the chance of 10\textsuperscript{th} International Conference for Nuclear Co-operation in Asia (ICNCA). The 1st ICNCA was held by the Atomic Energy Commission in March 1990 to promote co-operation in the field of nuclear energy with neighbouring Asian countries more efficiently. Since then, the Atomic Energy Commission of Japan has held many ICNCA conferences where the ministers in charge of development and utilization of nuclear energy exchanged frank views on how to proceed with regional co-operation, and has carried out practical co-operation on specified subjects as well. At the 10\textsuperscript{th} ICNCA Conference held in March 1999, it was agreed to move to a new framework, "Forum for Nuclear Co-operation in Asia" (including Co-ordinator and Project Leader System) with a view and information for shifting to more effective and organized co-operation activities. Under this framework, view and information for shifting to more effective and organized co-operation activities. Under this framework, view and information exchanges are made on the following 8 fields: (1) utilization of research reactors, (2) utilization of radioisotopes and radiation to agriculture, (3) application of radioisotopes and radiation for medical use, (4) public information of nuclear energy, (5) radioactive waste management (RWM), (6) safety culture of nuclear energy, (7) human resources development and new projects, and (8) industrial application of accelerators.

Nine countries are participating, i.e. Australia, China, Indonesia, the Republic of Korea, Malaysia, the Philippines, Thailand, Japan, and Viet Nam. The IAEA is invited as an observer. The basic framework of co-operation consists of the following three level activities:

- Ministerial Level Meeting (MM) and Senior Officials Meeting (SOM): Discussions are arranged on co-operation measures and nuclear-energy policies. MM and SOM are held once a year in the late autumn. The last MM and SOM were held in Hanoi, Viet Nam in December 2004, and the forthcoming MM and SOM will be held in Japan.
- Co-ordinators Meeting (CM): Discussions are performed on the introduction, revision and abolishment, adjustment, and evaluation of co-operation projects by an appointed co-ordinator from each country. CM is usually held in the beginning of March, and the venue is Japan.
- Co-operative activities for each project:
The present projects are as follows:
- Research Reactor Utilization
- Tc-99m Generator
- Neutron Activation Analysis
- Neutron Scattering
- Radioactive Waste Management
- Application for Agriculture
- Mutation Breeding
- Biofertilizer
- Nuclear Safety Culture
- Human Resources Development
- Public Information
- Application for Medical Care
- Radiation Oncology
- Industrial Application
- Low-Energy Accelerator.

2. Activities of radioactive waste management (RWM) project

2.1. Objectives

The objectives of this RWM Project are as follows:

a) management of safe control of radioactive waste, and
b) information exchange to enhance regional co-operation in the field of RWM.

The safe handling of radioactive waste is an important issue in a nuclear application development in FNCA countries in accordance with public enhanced interests to waste issues. Then, it was necessary to promote an international consensus on the safe management of radioactive wastes, to promote a regional co-operation in this field and to strengthen existing technologies among participating FNCA countries. The project was initiated in 1995 (at that time, under the ICNCA framework) to exchange information and to share experiences among the FNCA countries.

2.2. RWM Workshop

Annually, an RWM workshop is held at the host country where a technical visit is also held in conjunction with the workshop. During the workshop, topical issues are presented for a concentrated discussion. Recent themes of the sub-meeting were as follows:

Sub-Meeting theme at 2002 Workshop (Daejeon, Korea):
- Radioactive Waste from Decommissioning,
- Waste Characterization,
- Technologically Enhanced Naturally Occurring Radioactive Materials (TENORM).

Sub-Meeting theme at 2003 Workshop (Jakarta, Indonesia):
- Disposal of Low and Intermediate Level Waste (LILW) including waste acceptance criteria,
- Management of waste arising from decommissioning of small to medium scale nuclear facilities.

Sub-Meeting theme at 2004 Workshop (Kuala Lumpur, Malaysia):
- Regulatory aspects including exclusion, exemption and clearance on NORM/TENORM management,
- Waste treatment and characterization,
- Siting activities and safety assessment for disposal of LILW.

2.3. Outcome of RWM project

Major outcomes of this RWM Project until 2004 are following three reports:
(1) RWM Consolidated Report (FNCA RWM-R001):

The present status of radioactive waste management on FNCA countries is summarized in “the Consolidated Report”. This could engage one of the basic documents promoting the joint convention on Spent Fuel and Radioactive Waste Safe Management. Some countries still don’t join this convention, but this result will be one support of admission to this convention. Furthermore, this report will be a base of not only a mutual understanding of the preset status of radioactive waste management in FNCA countries but also a promotion of safety culture of radioactive waste management in each country.

(2) SRSM Activity Report (FNCA RWM-R002):

The SRS (Spent Radiation Source) management task gave us a grasp of present status of SRS management in member states and a direction to a solution of SRS good management. The resulting document will be a help of consideration of infrastructure promotion and regulation preparation.

The results of the Spent Radiation Source Management Task Group are summarized as follows:
- It was confirmed that the activities of this Spent Radiation Source Management Task Group were extremely significant, by exchanging frank views on ways to respond to problems technically and institutionally, and on any remaining issues, based upon practical experiences of the co-operative countries.
- It is important to discuss further on what the FNCA is able to do in deepening recognition on the importance of spent radiation source management in each FNCA country, and also in strengthening the safety awareness and management including public consensus.

This task also has a correlation with other projects, e.g. IAEA and Southeast Asia Management Program arranged by Australia. We can get a synergy effect by communication with each other.

(3) Activity Report of TENORM Task Group (FNCA RWM-R003):

The NORM/TENORM (Technologically Enhanced Naturally Occurring Radioactive Materials) task gave an impact for the recognition of NORM and the promotion for the preparation of regulatory guides.

This report was published in March 2005 and recommended to adopt a graded approach in its chapter 10, “Problems to be solved” Recommendations of this report are as follows:
[Step 1] The conceptual understanding is essential for a stable solution of NORM/TENORM problem.
[Step 2] Fact finding of NORM/TENORM based on the scientific data.
[Step 5] Indication of countermeasures and setting regulations if necessary.
[Step 6] Application to the real scene.

The reports are published and distributed to each FNCA country, and are also available in the FNCA web-site (http://www.fnca.or.jp/english).

2.4. Other activities

Regular Publication of Radioactive Waste Management Newsletter.

The RWM Newsletter is published twice a year. Publication alternates between Japan (odd-numbered editions) and the current Workshop host country (even-numbered editions). To date, editions No.1 to No.14 have been published in that manner and this agreed procedure will be continued. It should be noted that the articles are contributed from each FNCA country by voluntary base.
Updating of FNCA Maintenance of FNCA web-site (http://www.fnca.jp/english/).

The web-site, including summary report of workshops and RWM newsletters, reports such as RWM Consolidated Report and succeeding reports, has been enhanced.

All these activities (Reports, Journal Articles, RWM Newsletters, Web-site, etc.) are summarized to technical reports of FNCA and academic journals, and thus supply a basic knowledge to the relating scientists, technicians and governmental officers.

3. Conclusion

The continuous activities of FNCA-RWM gave us a development of management level on radioactive wastes in these nine countries of East Asia including Australia. We can expect a further expansion of these activities under harmonization with the IAEA.
National legal and regulatory infrastructure and implementation of the Code of Conduct in the Republic of Croatia

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Abstract. The Republic of Croatia adopted the radiation protection legislation from the former Yugoslavia and subsequently initiated its modifications to meet new circumstances. From the beginning, the requirements defined by the IAEA Basic Safety Standards (BSS) were considered and the legal solutions relied upon proposals given therein, taking, however, into account present infrastructure and actual circumstances. Information technology infrastructure and legal framework have been built which enabled the database managed by the State Office for Radiation Protection (SORP) to become base for all activities related to radiation sources control and authorisation. The Amendment of the Act on Protection against Ionizing Radiation was passed last year giving the SORP all required authorities except inspection. There is neither nuclear facility, nor production and depository of nuclear fuel or other radiation sources production in Croatia. As only one, temporary, storage for the spent sources is in operation in Croatia, and considering all problems arising when trying to site such a facility, the SORP has initiated an initiative on the regional level to find a solution for a joint depository. A new act is in preparation that will regulate import and export of radiation sources. Croatia is not a producer of sources so all the sources are imported. Any importer-user will be obliged to make a contract with the producer, by which the producer will re-import its source after the use.

1. Situation overview

In Croatia, 519 sealed sources in 75 institutions, 1503 x-ray units in 551 institutions and 9 accelerators are in use. 323 sealed sources are used in industry and similar sectors, out of which 113 are used for industrial radiography. Other sources are used in medical institutions and laboratories. X-ray units are used for medical diagnostics (753), in dentistry (528), veterinary medicine (27), luggage and shipment control (114) and industrial radiography (55). It is estimated that basically all sealed sources are registered and recorded in the SORP database. It is necessary to examine the sources that have not been used for a longer period of time and are not encompassed in regular annual surveillance checks.

Additionally, there are about 250 sources installed into lighting rods. Those are primarily Eu$^{152,154}$ and smaller number of Co$^{60}$ with activities ranging from 10 to 25 GBq. Legal obligation of the owners is to remove them by the end of this year. The situation is relatively difficult because the records about them are out-of-date and great number of sources is not regularly examined. The SORP has records on all sources installed in lighting rods, and their updating aimed at determining the present situation is in progress. The list is also compared to the list of sources disused and deposited into temporary storage. Special problem is that owners of all the sources are not known (some registered owners ceased to exist for various reasons, e.g. bankruptcy) or have no financial capacities to remove the sources installed in lighting rods as stipulated under the law.

Further, certain quantity of radioactive material (unsealed sources) is used in nuclear medicine and research laboratories. Total quantity imported during the last year was 6613.04 GBq.

There is neither nuclear facility, nor production and depository of nuclear fuel or other radiation sources production in Croatia. There are two temporary storages of radioactive material, one being still in use. It is properly organised and has accurate records on all the sources. About 300 sources deposited in other storage as well as disused sources kept in hospitals (150) and in industry (30) have to be taken care of.
Four reports related to orphan and lost sources were submitted since 2000:

- Year 2002 - 16 sources (Cs$^{137}$) were found in a small village. It was found out that they were used in a nearby factory (aluminium production) which had been closed decades ago.
- Year 2003 - a lightning rod was removed as a part of regular construction waste during the demolition of an old hotel in Dubrovnik and lost (never found).
- Year 2004 - smoke detectors were found after a building renovation.
- Year 2004 - a lightning rod was discovered by the Slovenian customs and returned to Croatia. Its origin was clarified.

2. Legislation

The Republic of Croatia adopted the radiation protection legislation from the former Yugoslavia and subsequently initiated its modifications to meet new circumstances. From the beginning, the requirements defined by the BSS were considered and the legal solutions relied upon proposals given therein, taking, however, into account present infrastructure and actual circumstances. The regulatory authority was traditionally the Ministry of Health (MH). In the first phase, the changes were not possible for various reasons. Therefore, as a temporary solution, with an intention to harmonise the system with requirements of the BSS as much as possible, the Croatian Radiation Protection Institute (CRPI) was founded as an institution subordinated to the MH. The CRPI was responsible for the keeping of the central national register of radioactive sources, users and exposed workers, for supplementary education, organization and supervision of other activities related to radiation protection and for providing expert assistance to the MH.

The CRPI activities were supported by the enacted Radiation Protection Act, which, among other, stipulated an obligation of reporting to the CRPI for all relevant organizations. The Amendment of the Act on Protection against Ionizing Radiation was passed last year giving the former CRPI, now the State Office for Radiation Protection (SORP), all required authorities except inspection. The regulations enacted cover the areas of authorization for use, surveillance, protection, transport, import and export of radiation sources. The areas of health surveillance and required qualification, including supplementary education for work with radiation sources are also covered. Intervals and methods for personal exposure monitoring, reporting and response in case of an accident have been defined.

The regulation on radioactive waste management and the emergency preparedness law are in the phase of preparation. The Act on Protection against Ionizing Radiation and the Amendment of the Act on Protection against Ionizing Radiation defines the State Office for Radiation Protection as an independent regulatory body and gives it authority as defined in the BSS, paragraphs 19 to 22 of the Code of Conduct (Code) and IAEA-TECDOC-1067. Inspection is still the competence of the Ministry of Health. A new Act is in preparation, by which the SORP will take over those tasks. By that change, the legal framework will be harmonised with the requirements of the Code and the BSS in all its main elements.

3. Infrastructure

The central institution in radiation protection is the State Office for Radiation Protection. It has been given the authority and the powers of a regulatory body. The SORP manages the central national register of radiation sources. Besides the basic data about sources, the data about equipment in which they are used, location, all surveillance examinations and decisions issued are kept. By connecting with the office management module, an overview of all sent/received documents related to a source is enabled. The SORP issues licenses for performing practices with radiation sources, approvals for procurement and licenses for use of radiation sources. To obtain a license, a user must submit a safety assessment and in some cases a security plan as defined in paragraph 20 of the Code. The license for use is renewed each year with a regular surveillance examination report submitted. The SORP may forbid use (suspend authorisation) and order removal of a source as well as request (according to the new law it will also conduct) inspection. The user is obliged to report the end of use of a source, indicate the disposal method and submit appropriate documentation.
The institution that manages temporary storage where spent or disused sources are deposited is obliged to inform the SORP of reception of any source as well as to send once a year a report with list of all deposited sources and their activities.

Three institutions are performing technical and expert tasks in the field of radiation protection, which include regular examination of sources and conditions for their use as well as dosimetric surveillance (IAEA-TECDOC-1067). They are authorised for the activities from paragraph 9 of the Code, except for intervention in the event of an accident or malicious act (competence of the SORP) and for calibration of radiation monitoring equipment. Related to the latter, activities on establishing a second standard laboratory have been initiated. It is expected to start operation by the middle of this year.

There are 23 institutions authorised for import, export and transportation of radiation sources. 47 institutions are authorised for health condition assessment.

Qualification of workers to perform tasks with radiation sources has been stipulated. Supplementary education, as well as periodic knowledge renewal and examination are conducted in the organisation of SORP in form of regular courses.

4. Spent and disused sources management

An approach to disused sources management includes resolving the present situation as well as creating provisions for efficient management in the future. Within the framework of the co-operation with IAEA and the Department of Energy of the American Government, Croatia has, through the activities of SORP, been included in the project of orphan sources searching, locating and identification. It has also been included in the project of the American Department of Defence and FBI, which is part of the international programme of radiation sources spreading prevention. Within the frame of those projects, the sources for which the owners are known but are not in use will also be collected and deposited into the temporary storage. As only one temporary storage for spent sources is in operation in Croatia, and considering all problems arising when trying to site such a facility, the SORP has initiated an initiative at the regional level to find a solution for a joint depository. A new act is in preparation, which will regulate import and export of radiation sources. Croatia is not a producer of sources and all the sources are imported. Any importer-user will be obliged to make a contract with the producer, by which the producer will re-import its source after the use.

Croatia has signed the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. The management and safe storage of spent or disused sealed sources project is in progress.
Radioactive waste management in Uzbekistan

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Abstract. The use of radiation sources and the management of radioactive wastes in Uzbekistan are outlined in this note. The respective national legal framework, current infrastructure, solid radioactive waste handling and the "Republic Radioactive Waste Disposal Facility" (RRWDF) are briefly described.

1. Introduction
Radiation sources and nuclear materials were used in Uzbekistan for a long time. All the radiation sources had been delivered from Russia. For storing and disposal of radioactive waste from users at that time, a disposal facility was established.

Radiation sources are widely used in the Republic of Uzbekistan for the successful development of technology, increase of yield in agriculture, and upgrade of methods and improvement of the quality of the treatment of patients. The main radiation sources are: Co-60, Cs-137, Am-241, Sc-90, Ir-192, Tc-99m, I-131, I-125. Moreover, there are irradiation facilities for industry, research and medicine (research reactor, accelerators, neutron sources), the use of which leads to the generation of radioactive waste.

2. Legal framework
All radioactive waste is managed under the legal framework as follows:

- Law of Radiation Safety;
- Law of Wastes;
- Law of Licensing of Separate Type of Activity;
- Sanitary Norms and Rules of Radiation Safety;

Supervision and control of the management of radioactive waste are carried out by interested ministries and agencies. A scheme of the governmental infrastructure relating to radioactive waste is given in Figure 1.

3. Management of radioactive wastes
Liquid wastes are collected at the workplaces, then mixed with pure water and released to the public water drain. But solid radioactive wastes are collected and sent to the Republic Radioactive Waste Disposal Facility (RRWDF).

In our country we have two groups of radioactive wastes, low level wastes and intermediate level wastes, for which we have special places to store and places for disposal. Radioactive wastes having more than 0.01 mSv/h up to 0.1 mSv/h are low level wastes and do not require shielding during normal handling and transportation. Radioactive wastes having a dose rate of 2 mSv/h are between low and intermediate level. According to national sanitary norms and rules of radioactive waste management, all disposal waste should have a surface dose rate of less than 0.01 mSv/h per package.
The main source of radioactive wastes is the Institute of Nuclear Physics, with Radiopreparat Enterprise, two cyclotrons, a gamma-source, radioisotope laboratory, and nuclear research reactor. For disposal of radioactive wastes to the RRWDF, all departments of the Institute prepared a special barrel with concrete shielding. This allows us to protect workers from exposure during storage and transportation. These barrels are sent to RRWDF for disposal. The republic radioactive waste disposal facility has enough special places and boxes, therefore we do not plan to close this facility.

![Diagram](image)

**FIG. 1: Governmental infrastructure relating to radioactive waste.**

The Republic of Uzbekistan has three types of storage and disposal facilities: local storage places, local disposal places of the institutions and organizations, and the Republic Radioactive Waste Disposal Facility. All these places and the RRWDF use close surface places or special buildings.

RRWDF, established in 1959, is located 35 km to the East direction from Tashkent in the mountain place. This facility has two tanks for liquid waste, one building with 16 cells of 5 m depth near the surface for solid waste and four well-type cells of 4 m depth for intermediate waste. RRWDF has special boxes for decontamination of cars and personnel after disposal of the delivered radioactive waste. The facility has a full infrastructure, staff, cars and spaces. RRWDF disposes all radioactive waste from medicine, industry, research institutions and agriculture.

Each institution and organization has a member of its staff, or a department, responsible for radiation protection and safety at these places. Under their control, the organization collects radioactive waste in a special place and once a week, or once a month, it sends radioactive waste to RRWDF. RRWDF has a special car for collection and transportation of radioactive waste from all regions of the country. For intermediate level radioactive waste there are specially prepared concrete packages of metal flanks for decreasing the radiation level.
Central radioactive waste management facility

_The existing storage facility in the United Republic of Tanzania_

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**Abstract.** In the United Republic of Tanzania radioactive wastes are generated from various practices that include medicine, industrial application, education, research, agriculture, exploration, and also from illicit trafficking and orphan sources. Adequate management of radioactive wastes is required to ensure the protection of human health and the environment without imposing undue burden to the future generations. There are several procedures in the management of generated radioactive wastes which are categorized by three steps comprising handling, treatment and conditioning processes before disposal. To carry out all these procedures a proper storage facility is necessary. Unfortunately not all developing countries, Tanzania being one of them, have disposal facilities particularly for high-level waste due to associated high costs. Therefore, interim storage facilities are seen as an alternative option. Such facilities can be designated areas, rooms or buildings or sites dedicated to the storage of radioactive waste. In any such case, the waste will be placed in the store with the intent of later being retrieved. In Tanzania the radioactive wastes are stored in the Temporary Radioactive Waste Management Facility (TRWMF) before being transferred to the new Central Radioactive Wastes Management Facility (CRWMF), which is in commissioning stage. The CRWMF is constructed in such a way that the access for later retrieving radioactive wastes is easily allowed once the disposal facility is available. This paper reviews safety procedures followed during siting and design of CRWMF, foreseen operation procedures, waste package acceptance criteria and the security upgrades of the facility.

1. **Introduction**

Radioactive waste as a source of ionizing radiation represents potential hazards to human health and must be carefully managed so as to reduce associated risks to acceptable levels [1]. The United Republic of Tanzania, in October 1999, with the assistance of an IAEA recruited expert from South Africa succeeded to condition radium needles from teletherapy units. The storage of this radioactive waste raised concern and commencement of collection of other disused/spent radioactive sources from users prompted the need for having a central facility in the Country to manage all types of radioactive waste. A Temporary Radioactive Waste Management Facility (TRWMF) had been constructed and a modified metal shipping container was acquired to put the radioactive waste safely and securely. This facility was not an adequate storage facility to accept high level waste and waste neutron sources, and could not accommodate all wastes generated in the country due to its size. In the year 2000, the Tanzania Atomic Energy Commission (TAEC) decided to construct another facility, CRWMF, to serve the purpose at a Commission’s new site. The new facility that is at commissioning stage has been made from concrete blocks, and has a smooth floor and walls to allow easy decontamination. The facility comprises office, laboratory and septic tanks used as receipt of water from treatment of waste. This paper reviews safety procedures followed during siting and design of CRWMF, foreseen operation procedures, waste package acceptance criteria and the security upgrades of the facility.

2. **Siting**

During the site selection for the CRWMF, attention was paid to safe operation during the service life of the facility. Based on safety principles, key factors for siting, including geology, hydro-geology, meteorology, land use and demography, were considered to ensure that human health and the environment are protected. This criterion was defined by the national regulatory authority and other international agencies to fulfil the maximum safety during the operational life of the facility. Unfortunately geological and meteorological data at the area were not collected during siting due to
the costs associated with the collection of data. Nevertheless, historical data were used for selecting
the site. The location of the facility is outside the town of Arusha, in an under populated area with no
running and stationary waters nearby the facility. There are no national parks or areas with historical
monuments or archaeological findings near the facility. The area where the facility is located has no
record of seismic events. However, there are dormant and dead volcanic mountains, Mount
Kilimanjaro and Mout Meru, situated about 105 km and 40 km respectively from the CRWMF site and
this fact cannot guarantee permanent safety from earthquake events. Wind and rainfall in the area are
moderate throughout the year and free from snow cover as the site is situated in a tropical zone. There
is also no history of storms in the area. It is strongly recommended to the management of the site to be
in close contact with the meteorological agency to know the weather forecast so that necessary
measures can be taken to keep the facility in a safe way in case of a disaster. There are no hazardous
installations such as oil refineries, chemical plants, storage depots, pipelines and other facilities that
could have an impact on the facility operation. Transportation routes with frequent movement of
hazardous materials such as fuel and industrial gas are about 24 km away from the facility. The town
planning authority was made aware that the site should not be allocated for building industries which
would endanger the operation of the facility. The ambient background radiation survey at the area was
recorded as 0.2 µSv/h.

3. Design

The design and operation of storage facilities for radioactive waste require to provide for the protection
of workers and members of the public, and for the protection of the environment [2]. The CRWMF has
been designed to provide adequate shielding for workers and the members of the public. The thickness
of the walls of the facility is 30cm and the thickness of the doors in storage rooms is 3.2 cm, 1.2 cm,
0.6 cm and 0.3 cm of steel sheet respectively (see Fig. 1). The shielding effectiveness was calculated
such that the doses for workers and members of the public are within acceptable limit. As the CRWMF
is located in a tropical zone, where there is normally high temperature and humidity, the building has
been constructed with enough ventilation at all sides of the roof to control the situation. Genuine
electrical devices are installed in the facility, as planned in the design, to reduce the probability of fire
in the facility. Concrete septic tanks were incorporated in the facility to collect water from handling
activities and it was suggested that during operations that working benches and on floors be covered
with aluminium sheets to prevent contamination. The installation of the septic tank took into
consideration IAEA Basic Safety Series No.115, which recommends that any discharge to the
environment, and exposure to workers and the general public, should be controlled and limited [3].
Also, a standby generator has to be placed in the facility as power backup in case of loss of electricity.
The safety devices like fire extinguishers and smoke detectors are to be installed. In addition, security
systems have been installed which comprise radio communication system, infrared motion sensors,
and magnetic balance switches on all entrances.

4. Operating procedures

The requirements for the radiological protection and safety of members of public and workers at the
repository are similar to those applicable to other operating facilities in which radioactive materials are
being handled [1]. Based on the aforementioned requirement, the operation procedure for the facility
has been designed to meet requirements established by IAEA guidelines dealing with the management
of radioactive waste. During the commencement of operation of the facility, operational data
monitoring of radiation levels will continue near the facility and any contribution of doses or increase
of doses in the vicinity of the facility will be recorded and documented. Any dose differences from
ambient background radiation in the vicinity of the facility should be identified and used to raise the
need to review the design of the facility or extra radiation safety precautions to be implemented to
protect the members of the public. Any liquid released from handling/treatment of waste will be
retained in concrete septic tanks constructed in the facility for analysis before released to the
environment in accordance with the of limit provided in the Atomic Energy Regulations 2004 [4]. It
has been agreed by the management of the facility that, before the facility operates, essential tools and
devices used in the facility should be available. These include a telephone network, mobile shield,
protective gears such as lead aprons, hand lead gloves and lead glasses, a set of lead bricks, portable or
mobile dose rates meters for dose monitoring, radon detector, fume hood, remote tongs, hand pallets tracks, heavy duty pallets for easy moving of the heavy containers and machines, and a crane for the safe handling of waste packages or damaged waste packages. The fixed or portable instruments to detect external contamination will be provided at exits of the controlled areas. All monitoring equipment will be periodically tested and calibrated. Procedures will be available for waste packages inwards/outwards, inventory control and reporting, responsibilities, effective management of security systems, dealing with emergencies at the facility and restriction on entries to the facility. Sufficient and well-trained personnel should be given priority for the proper and safe operation of the facility.

A, B, C, D and E are doors of 0.3cm, 3.2cm, 1.2cm, 0.6cm and 0.3cm respectively
\[ t = \text{thickness of wall } 30\text{cm} \]
\[ = \text{barrier} \]

**FIG. 1. CRWMF Layout**

5. **Waste packages acceptance criteria**

Waste package acceptance in the facility will be based on the following criteria:

(i) each package should be identified and characterized to ensure safe handling, storage, accountability and activity control,
(ii) waste packages should be capable of withstanding damage in any accident,
(iii) any packages or materials that generate gas, heat, corrosion and combustibility or cause activation will not be accepted to the facility except modifications are done,
(iv) external dose rates and surface contamination of the waste packages received in CRWMF should comply with IAEA transport requirements [5].
6. Security system of CRWMF

Infrared motion sensors, balance magnetic switch, doors made from steel sheets, multiple T-locks and interlocks, fences and communication system to monitor unauthorized entries to the facility have been provided to enhance the security system of the CRWMF. The procedure for security operations in the facility involves training security guards on site. In addition to the adequate communication system, key control procedures and reporting of incidents to the relevant authorities, including the regulatory authority, police and any relevant authority have been arranged. In the context of upgrades and monitoring of security, the CRWMF has implemented measures as recommended by the IAEA [6].

7. Conclusion

Safety and security of radioactive waste is currently a world-wide concern, following the possibility for diversion of the radioactive materials for malicious use. Tanzania, being aware of this danger, has taken necessary steps, including commissioning of the CRWMF, to keep all spent and orphan sources safe and secure. In this paper the authors highlighted steps taken in siting, design and operation, respective procedures, waste packages acceptance criteria and the security system of the facility.

REFERENCES

Current regulatory status of Turkey on radioactive wastes arising from nuclear installations

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Abstract: General information about the existing and planned regulatory structure concerning radioactive wastes arising from nuclear installations in Turkey is given in this paper. Radioactive wastes should be regulated to ensure the protection of human health and environment now and in the future. Although Turkey doesn’t have a NPP there are two research reactors and a fuel pilot facility at the moment, so it is necessary to have well-developed regulatory structure in other words regulatory arrangements related to the radioactive wastes arising from nuclear facilities should be in place.

1. Introduction
Although Turkey doesn’t have a nuclear power plant for the moment, nuclear energy may seem to be one of the alternative energy options of the country in the near future. However there are two research reactors and a pilot fuel facility for the time being. Turkish Atomic Energy Authority (TAEA) is the nuclear regulatory agency of Turkey. Nuclear Safety Department (NSD) is one of the departments of TAEA and responsible for licensing, developing regulations, carrying out inspections related to siting, construction, operation and decommissioning of nuclear installations. NSD is also responsible for regulating on-site management of radioactive wastes arising from nuclear installations, since on-site radioactive waste management activities are associated with the nuclear power production, nuclear research and nuclear fuel cycle practices.

1.1. General outlook to nuclear facilities in Turkey
There is no NPP in operation, under construction, or decommissioned in Turkey. However studies to build a NPP in Turkey was started in 1965. Akkuyu site was licensed by TEAE in 1976. The last nuclear power plant construction project which would be Turkey's first nuclear power plant has been delayed in 2000 due to some financial and political issues.

The 250 kW TRIGA Mark-II research reactor, one of the existing nuclear facilities in the country, is located in Istanbul Technical University in Istanbul. The facility is in operation since 1979.

The 5 MW TR-2 research reactor is located in Çekmece Nuclear Research and Training Centre in Istanbul. The facility is in re-licensing stage, has been given 300 kW limited operation permit. 1 MW TR-1 research reactor was decommissioned at the end of 1970s, a sort of re-powering decommissioning strategy was adopted and TR-2 reactor has been located in the pool of TR-1.

The pilot fuel facility is located in Çekmece Nuclear Research and Training Centre. The plant which was established in 1985-86 and is in licensing stage right now. The uranium purification, uranium pellet fabrication studies, uranium-thorium mixed oxide studies, analyses of nuclear fuel design and performance are carried in this facility.

There is also a radioactive waste processing plant at the Çekmece Nuclear Research and Training Centre. This facility processes the radioactive wastes and spent radioactive sources coming from medicine and several industries and also TR-2 research reactor. The wastes are processed using chemical techniques and those containing radioactivity level below the drinking water standards are discharged to the sewer, those above this level are compacted, cemented and stored in this storage. This facility is treated as a radiation facility rather than a nuclear facility.
2. Regulatory status of radioactive wastes arising from nuclear facilities

2.1. Licensing procedure related to the radioactive waste management facilities

Licensing procedure of nuclear installations is defined via Decree Pertaining to the Licensing of Nuclear Installations which was issued in 1983 [1]. According to this decree the radioactive waste management facilities including radioactive waste disposal facilities are one of the nuclear fuel cycle facilities which should be licensed in different stages.

These facilities are firstly given site license after approval of site report the content of which is determined by TAEA. For getting construction license, the applicant should have site license and should submit preliminary safety analysis report to the regulatory body, TAEA and get approval. Construction license is issued by two steps, which are limited work authorisation and construction license. Finally as steps for operating license, permits for test operations and full capacity operation are issued.

According to draft Nuclear Act there are licensing arrangements related to the radioactive waste management facilities including disposal facilities such that the radioactive waste management facilities including disposal facilities shall be licensed by TAEA for operation, also, when necessary, their site evaluation, construction, commissioning, modification in terms of safety or security aspects and closure may subject to different permits issued by TAEA [2].

2.2. Legislation for radioactive wastes

The Turkish Atomic Energy Authority Law is a sort of law that defines the duties and responsibilities of the departments within TEAE and TEAE itself in a brief and general manner [3]. In this law, it is stated that TAEA is responsible for taking all necessary measures to ensure safe management including handling, transport, processing, storing and disposal of all wastes arising from nuclear installations and Nuclear Safety Department is authorised to ensure radiation protection and environmental safety during whole licensing stages of nuclear installations.

The responsibility of on-site radioactive waste management activities for the NPPs and research reactors rests with the licensee of the related practice according the following draft regulations.

The radioactive waste management is one of the basic technical principles defined in the draft regulation called “Basic Principles for the Safety of Nuclear Installations”. The radioactive wastes arising from nuclear facilities should be kept as low as reasonably achievable both in volume and activity. The operating organization of the nuclear installations should develop and implement a radioactive waste management program for handling, processing, transport and disposal of the wastes in a safe manner. For the decommissioning phase the decommissioning program which should ensure the exposure of radioactive materials is kept as low as reasonable achievable should be submitted to the regulatory body, after approval the decommissioning activities can be started [2].

According to the draft regulation called “Specific Principles for the Safety of Nuclear Power Plants” the specific principles related to the siting, design, construction, commissioning, operation and decommissioning stages of the NPPs are defined. There are some arrangements related to the radioactive wastes in this document. The radioactive waste management systems shall be designed such that the care is taken to provide for conservative adherence to authorized limits. The design should ensure that all plant components containing radioactive materials are adequately shielded and that the radioactive material is suitably contained. Design should take into account radiation protection requirements with the attention to the shielding requirements, confinement of radioactive materials, accessibility, access control, the need for monitoring and control of the working environment and decontamination. Design should ensure that the personnel are adequately protected against the potential radiological hazards in the course of on-site storage activities of the spent fuel. The spent fuels should be subcritically configured and be safe under all storage conditions. Shielding and safe methods for loading to the transportation cask should be provided. The design should provide redundant and reliable systems for residual heat removal to protect the integrity of the cladding of the spent fuels. Design should also provide with conditions for the inspection of the spent fuels and
detection of failures within the spent fuel. Measures should be taken at the design and operation so that the radioactive wastes can be kept as low as reasonably achievable during the decommissioning phase [2].

Draft regulation titled “Specific Principles for the Safety of Research Reactors” defines the specific principles related to design, construction, commissioning, operation and decommissioning stages of the research reactors. Related to the radioactive wastes there are some arrangements. The design of radioactive waste systems shall have adequate provisions for control and monitoring of radioactive effluents and for handling, processing, transport, storage and disposal of radioactive wastes. The procedures including monitoring and recording of radioactive wastes shall be implemented during operation stage [2].

The draft regulation called “The Format and Content of Safety Analysis Report for Research Reactors” defines the information and analyses which should be supplied by the applicant to the regulatory body for the research reactors to grant the operation license. Applicant shall give the information of the origin, radioisotope composition, the predicted annual quantities together with the estimation methods of the solid, liquid and gaseous radioactive wastes and those stored and disposed in or off site. The environmental discharge and transport conditions from the facility site of the wastes, the methods of monitoring, minimization and segregation from the non-radioactive wastes should be explained and limiting conditions of operation related to the wastes should be derived [2].

In accordance with the draft regulation titled “Recording and Reporting Requirements for Research Reactors” the operating organisation of the research reactors shall record the amount, volume, the class (high, intermediate, low), radioisotope composition, physical and chemical characteristics and the origin of all solid, liquid and gaseous radioactive and nuclear wastes and those stored or disposed in or off site. These records should be kept through the life of the reactor. Once a year the reports of the information mentioned should be submitted to the TAEA [2].

According to draft Nuclear Act “radioactive waste” is defined as the radioactive material or equipment/ structure that are contaminated with radioactivity at concentrations or activities greater than clearance levels defined by TAEA and for which no further use is foreseen. There is also a definition for “radioactive waste management” that is all activities that are involved in handling, processing, storage and disposal of radioactive wastes. The responsibility of on-site radioactive waste management activities associated with nuclear power, research reactors or fuel cycle facilities rests with the licensee of the facility. Whoever is responsible for the generation of the radioactive waste should bear the responsibility for the cost of its management including disposal. “National Radioactive Waste Management Fund” shall be set in that the licensee shall contribute annually to that fund management of which is Ministry of Treasure is responsible for. The means and the amount of the contribution will be defined by TAEA. In addition the import, storage and disposal of radioactive wastes that are not generated in the country are not allowed in the territory Turkey [2].

3. Regulatory aims for future work related to the radioactive wastes

In order to ensure steady and stable development of nuclear research and fuel cycle activities and aware that there is a possibility of the construction of nuclear power plant in Turkey the main regulatory aim for radioactive waste management is to have a well-structured regulatory framework. Future plans are of short and long term, focussing in the short term on:

- radioactive waste management programme for NPP;
- clearance levels for the radioisotopes from nuclear fuel cycle activities;

and in the long term on:

- regulations to ensure the safe decommissioning of nuclear installations.
The radioactive waste management programme for nuclear power plants started being drafted as a safety related document for licensing of NPPs and it is considered to include the issues of safe management for radioactive wastes generated over the lifetime of the plant under normal and accident conditions.

Clearance levels in terms of dose or activity concentration for the radioisotopes from nuclear fuel cycle activities are being planned to set forth in the short term. Clearance levels of the radioactive wastes from the usage of radioactive materials have already been defined in the Regulation on the Radioactive Wastes from The Usage of Radioactive Materials which was issued in 2004 [4].

Even though decommissioning is not an urgent issue in terms of licensees and TAEA for the time being, Turkey having 3 nuclear facilities will face this issue in the future like the other countries. The format and content of decommissioning plan, the regulation for funding mechanisms etc. are being planned to develop for the long term.

REFERENCES

Disposal: a last step towards an integrated waste management system in Egypt

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Abstract. Radioactive wastes in Egypt have been generated from activities associated with peaceful application of nuclear technology. The development of a safe radioactive waste management is one of the critical issues for the future of nuclear applications. The handling, transportation, treatment, conditioning, storage, and disposal of radioactive waste should be performed with the objective of achieving adequate safety and protection of both the human health and the environment. This goal forms the basis of the waste management policy in Egypt. This paper is concerned with the main radioactive waste management policy principles that are implemented through the atomic energy authority of Egypt and other end users. The Egyptian radioactive waste management system is presented with special reference to the disposal element including site selection studies, engineering design, and performance assessment studies. The ultimate aim of these assessments is to evaluate the performance of the disposal site against individual dose or maximum permissible concentration as a regulatory measure. In this respect, the bathtub and groundwater scenarios are considered during two phases of the disposal site namely; license application and waste acceptance criteria.

1. Introduction

The Radioactive waste management policy serves as a national commitment to address the radioactive waste issues in the country. The Egyptian radioactive waste management policy is a framework to interact with the world and our past, present and future. In addition to the internationally accepted principles developed by the International Atomic Energy Agency (IAEA), Egypt has its own policy principles. The international principles are applicable to all countries and can be applied to all types of radioactive waste, regardless of its physical and chemical characteristics or origin. These sets are the protection of human health and the environment, now and in the future without imposing undue burden on future generation [1]. The national policy principles are national legal framework, control of radioactive waste generation, safety of facilities, waste generator pays, sound decision-making based on scientific information, risk analysis and optimization of resources, precautionary principle and finally International co-operation [2].

The national legal framework provides laws, regulations and guidelines for radioactive waste management taking into account the overall national radioactive waste management strategies. Generation of radioactive waste is kept to the minimum practicable, in terms of both its activity and volume. During all radioactive waste management activities priority is given to safety matters including the prevention of accidents and limitation of consequences if accidents occur. The financial burden for the management of radioactive waste is borne by the generator of that waste. Decision-making is based on proven scientific information and recommendation of competent national and international institutions dealing with radioactive waste management. Where there is uncertainty about the safety of an activity a conservative approach shall be adopted. The government recognizes that it shares a responsibility with other countries for global and regional radioactive waste management issues. Our actions follow the principles of this policy and all relevant regional and international agreements.

Over the past twenty years, the waste management centre at Inshas has begun to build an integrated waste management system to serve the Egyptian nuclear activities based on the above policy principles. During the establishment of the Egyptian waste management system, a range of technological options have been identified and evaluated in order to select and justify the most appropriate solution by taking into account the basic waste management principles, regulatory
requirements, resources available, and transport regulations. The selection of a technology for each element in the system has been bounded up with the selection of the overall system for safe management of the radioactive waste. Fig. 1 illustrates the basic elements in the Egyptian radioactive waste management system.

FIG. 1: Radioactive Waste System In Egypt

2. Radioactive waste management system

During the establishment of the waste management system, all stages in waste processing have been considered, starting from waste generation, through sorting and treatment until disposal of these wastes. To achieve the overall safety goal of waste management, component elements must be complementary and compatible with each other. The core of the waste management system is the technology which is applied to the waste from its generation to its disposal.

2.1. Waste generation

Over the past four decades, the peaceful applications of nuclear technology in research, medical and industrial purposes have generated a wide range of radioactive wastes. The classification of the generated waste follow the international classification proposed by the IAEA, which categorize the waste to low-level waste, intermediate level waste, high level waste and sealed sources [3]. The following waste management elements deal with the first two categories.

2.2. Transportation

Safety in the transport of radioactive material is provided through transport regulations, which aims to protect persons, property and the environment from the effects of radiation during the transport of these materials. The transport regulations include certain requirement on the waste package to survive accident conditions i.e. drops of waste packages from a high-speed truck.

2.3. Treatment

Treatment includes operations that reduce the volume of the generated wastes. There are various volume reduction technologies; the selection of any of these technologies is largely depending on the waste type. The radioactive waste generated in Egypt can be classified as aqueous waste, organic liquid waste, and solid waste.

2.3.1. Aqueous waste

The treatment of these wastes aim to split the waste into small stream of concentrate containing the bulk of radionuclides and a large stream its contamination level is sufficiently low to permit its discharge to the environment. During the selection of the aqueous waste treatment technology, a set of
decisions factors has been discussed. These factors include the characterization of generated waste, discharge requirements for decontaminated liquors, available technologies and their costs, conditioning of concentrates resulting from the treatment, and storage and disposal of the conditioned concentrates [4]. Based on the above factors, the selected technological option is illustrated in Fig. 2.

![Treatment of Aqueous Waste](image)

**FIG. 2: Treatment of Aqueous Waste**

### 2.3.2. Organic liquid waste

Despite the smallness of the generated organic liquid waste volumes by comparison with aqueous radioactive waste, there is a need for effective waste treatment technology that can minimize the detrimental effects on health and the environment. The ‘dilute and disperse’ option open for some aqueous and gaseous waste is not appropriate for most of organic liquid waste. The incineration option has been selected, the aim from this step was to achieve a complete combustion of the waste to inorganic products that can be easily handled in subsequent elements.

### 2.3.3. Solid waste

The incineration and compaction have been selected in Egypt to deal with the solid waste. Incineration as in the case of organic liquid waste, has been considered after careful evaluation of all features, especially radiological aspects [5]. Compaction involves compressing the waste into containers or boxes in order to reduce the volume.

### 2.4. Conditioning

Conditioning includes operations that produce a waste package suitable for handling, transportation, storage and disposal [6]. The immobilization of radioactive waste to obtain a stable waste form may be the most important step to minimize the potential for migration or dispersion of radionuclides into the environment during storage, handling, transportation and disposal. The choice of the immobilization in a cement matrix has been based on the physical and chemical nature of the waste, low cost, suitability for sludge, good thermal, chemical, and physical stability, and good compressive strength of the waste forms. Then the immobilized waste is packed, the waste packages must be capable of meeting shielding and containment requirements for handling, storage, transportation and finally the waste disposal site requirements.

### 2.5. Storage

Storage is an integral part of the waste management system. The main functions of a storage facility are to provide safe custody of the waste packages and to protect both operators and the public from
any radiological hazards associated with radioactive waste. The design of storage facilities has met the national regulatory standards and basic safety principles. The interim storage facilities have been designed to facilitate inspection and monitoring of stored waste, keep exposure to personnel as low as reasonably achievable principle (ALARA), and provide adequate environmental conditions to ensure proper conservation of waste packages during their tenure at the facility. Also to keep record and identification of stored waste packages [7].

2.6. Disposal

The basic objective of disposal is to isolate the waste from water and the human environment under controlled conditions for long time to allow the radioactivity to either decay naturally or slowly disperse to an acceptable level. The choice of a disposal option depends on the waste type and local conditions, including geological and hydro-geological conditions, radiological performance requirements, and considerations of socio-political acceptance. According to the international standards, Egypt selects the shallow land disposal option to dispose low-level radioactive wastes.

3. Radioactive waste disposal assessment

Since disposal is the final step in any radioactive waste management system, it has been the subject of a number of studies, evaluations, and reviews. These included the metrological, geophysical, geological and hydrological studies, and total performance assessment for the disposal system.

3.1. Site selection studies

An important aim of site selection studies is to select the most convenient site to host the disposal facility and to provide the data required for the development, evaluation and use of the conceptual models and corresponding mathematical models required for a quantitative assessment of long-term performance. These studies have been performed to survey the site characteristics including hydrogeology, geochemistry, tectonics and seismicity, surface processes, and meteorology [8].

The geophysical studies conducted to explore the aquifer beneath the compass of the atomic energy authority, investigate the lithology and structural fabric of the area and to trace the direction of flow within these aquifers. The results of the vertical electrical receptivity led to the recommendation that the eastern plateau is the most convenient place to host the facility because of its formation from impervious limestone rock that prevent the groundwater circulation beneath the facility [9].

3.2. Engineering design

The disposal facility should be designed to increase site stability, inhibit inadvertent human intrusion, retard radionuclides releases, and protect the waste package from some of the initiating events. The engineering design should minimize the need of maintenance after closure and take into account the uplift pressure due to the water table, loading due to the disposal of the waste, and shielding.

The Egyptian repository is comprised of four modules (3x5x10m each) with capacity of 6000 concrete containers. The water movement is controlled by drainage system link the four modules to drain the precipitation away from the vault surface. The facility has fully engineered structure: backfill material reinforced concrete walls, and multi-layer cover. The cover system is designed to minimize water infiltration, to direct percolating or surface water away from the disposed waste, and to resist degradation by surface geologic processes and biotic activity.

3.3. Performance assessments studies

Performance assessment is a tool for the regulatory body that is used in performing confirmatory analyses in support of license application. It was recommended that the modular approach is used in conducting the performance assessment to support a proposed disposal facility. The modular structure allows the use of very simple models, updates as better models are developed and more site-specific information is obtained. In this respect, two scenarios were considered during two phases of the disposal namely; application of construction licence and development of waste acceptance criteria. During the application for the construction licence a bathtub scenario was selected to predict the performance of the total disposal system. Where during the development of waste acceptance criteria a groundwater scenario was considered.
3.3.1. Bathtubbing Scenario

The selection of this scenario based on the precautionary principle, it begins with infiltration from flood, precipitation or run-off entering the disposal unit. Dissolution of radionuclides begins and leachate accumulates in the disposal unit, like a bathtub. This occurs due to the low permeability of the surrounding formations. The disposal unit eventually fills and overflows. Human exposure may result from the ingestion of the contaminated water. Based on this scenario, the transport of radionuclides has been performed for saturated zone. The Advection Depressive Equation (ADE) that govern radionuclides migration through the saturated zone is as follow:

\[
\frac{\partial R_d C_i}{\partial t} = -u \frac{\partial C_i}{\partial x} + D \frac{\partial^2 C_i}{\partial x^2} - \lambda C_i \quad (1)
\]

\[
R_d = 1 + \rho (1 - \varepsilon) K_d / \theta
\]

Where

- \( C_i \): concentration of radionuclide i, Bq/l
- \( u \): velocity of flow, m/y
- \( D \): dispersion coefficient, m\(^2\)/y
- \( \lambda \): decay constant, y\(^{-1}\)
- \( \theta \): moisture content in the facility, m\(^3\)/m\(^3\)
- \( K_d \): distribution coefficient of the radionuclide on backfill matrix,
- \( \rho \): density of the backfill material, kg/m\(^3\)
- \( \varepsilon \): porosity of the disposal unit, and

The release rate is given by:

\[
Q_i = C_i \cdot v \cdot A
\]

Where

- \( Q_i \): release rate, Bq/year, and
- \( A \): horizontal area of the disposal module, m\(^2\).

The dose rate arising from drinking contaminated groundwater from a well is given by:

\[
D_{\text{sw}} = E C_i \cdot Q_i \cdot DCF_i
\]

Where

- \( D_{\text{sw}} \): effective dose from drinking water, mrem/y
- \( E \): empirical constant equal \( 3.7 \times 10^3 \),
- \( DCF_i \): Dose Conversion Factor mrem/Bq
- \( Q_i \): total release rate into the rock volume affected by the well (Bq/y),

The PAGAN code has been selected to run simulations for Cs-137, Co-60 and Sr-90 radionuclides transport from the disposal facility to a well located 150 m away from the disposal site. The inventories and properties of these radionuclides are shown in Table 1 and 2. The results of the simulations are presented in Table 3. It is clear from these results that the total effective dose resulting from drinking two liters daily of ground water from a 150 m well from the disposal facility is lower than the permissible (25 mrem/year) for public [10].

<table>
<thead>
<tr>
<th>Property</th>
<th>Value</th>
<th>Unit</th>
</tr>
</thead>
<tbody>
<tr>
<td>Porosity</td>
<td>0.3</td>
<td>unitless</td>
</tr>
<tr>
<td>Dispersion Coefficient</td>
<td>0.27E+03</td>
<td>m(^2)/year</td>
</tr>
<tr>
<td>Velocity</td>
<td>0.54E+02</td>
<td>m/year</td>
</tr>
<tr>
<td>Density</td>
<td>1.81E+3</td>
<td>kg/m(^3)</td>
</tr>
<tr>
<td>Horizontal area of the disposal</td>
<td>200</td>
<td>m(^2)</td>
</tr>
</tbody>
</table>
Table 2. Properties Of Radionuclides Used In The Modelling

<table>
<thead>
<tr>
<th></th>
<th>K_d (m^3/kg)x 10^-3</th>
<th>Inventory (Ci)</th>
<th>DCF (Sv/Bq)</th>
<th>Half-life (year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cs-137</td>
<td>112.00</td>
<td>600</td>
<td>1.4*10^-8</td>
<td>3.01E+1</td>
</tr>
<tr>
<td>Sr-90</td>
<td>075.00</td>
<td>100</td>
<td>2.8*10^-8</td>
<td>2.81E+1</td>
</tr>
<tr>
<td>Co-60</td>
<td>036.74</td>
<td>400</td>
<td>9.2*10^-10</td>
<td>5.25E+0</td>
</tr>
</tbody>
</table>

Table 3 The Source Rate and Total Effective Dose in Groundwater.

<table>
<thead>
<tr>
<th>Source Rate (Ci/year)</th>
<th>Total Dose Rate (mrem/year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cs-137</td>
<td></td>
</tr>
<tr>
<td>After 10 years</td>
<td>2.38*10^-1</td>
</tr>
<tr>
<td>After 300 years</td>
<td>3.31*10^-4</td>
</tr>
<tr>
<td>Sr-90</td>
<td></td>
</tr>
<tr>
<td>Co-60</td>
<td></td>
</tr>
<tr>
<td>After 10 years</td>
<td>6.01*10^-2</td>
</tr>
<tr>
<td>After 300 years</td>
<td>4.6*10^-5</td>
</tr>
<tr>
<td></td>
<td>~ 0</td>
</tr>
<tr>
<td></td>
<td>3.37*10^-10</td>
</tr>
</tbody>
</table>

3.3.2. Groundwater scenario

The groundwater scenario begins with precipitation or run-off infiltrates through the disposal cover. Radionuclides leach from waste forms and are transported downward by gravity through the unsaturated zone to the water table. Based on this scenario, the performance of the unsaturated zone as a natural barrier under different initial flow conditions has been evaluated by conducting a numerical simulation for pulse and constant source of Tc-99 injected to the unsaturated zone. The ADE that govern the transport through the unsaturated zone is as follow:

\[
\frac{\partial R_{d\text{C}_i}}{\partial t} = \frac{\partial}{\partial x} \left[ \theta \left( D_{xx} \frac{\partial C_i}{\partial x} + D_{xz} \frac{\partial C_i}{\partial z} - C_i V_x \right) \right] + \frac{\partial}{\partial z} \left[ \theta \left( D_{xz} \frac{\partial C_i}{\partial x} + D_{zz} \frac{\partial C_i}{\partial z} - C_i V_z \right) \right] + I \tag{3}
\]

Where

\( \theta \) volumetric water content, cm^3/cm^3

\( I \) source/sink term

The velocities and the dispersion coefficients component are given by:

\[
V_x = -K(\psi) \frac{\partial \psi}{\partial x} \quad V_z = -K(\psi) \left( \frac{\partial \psi}{\partial z} + 1 \right)
\]

\[
D_{xx} = \alpha_i \frac{V_x^2}{V} + \alpha_t \frac{V_x^2}{V} + D_o \alpha e^{b_0} \quad D_{zz} = \alpha_i \frac{V_z^2}{V} + \alpha_t \frac{V_z^2}{V} + D_o \alpha e^{b_0}
\]

\[
D_{xz} = (\alpha_i - \alpha_t) \frac{V_x V_z}{V} \tag{4}
\]

Where

\( D_o \) diffusion coefficient of solute molecules in free water, cm^2/hr

\( a, b \) tortuosity constant

\( \alpha_i, \alpha_t \) longitudinal and transverse dispersivity.
The volumetric water content is given by Richard equation as follows:

\[
\frac{\xi}{\partial t} = \frac{\partial}{\partial x} \left[ k(\psi) \frac{\partial \psi}{\partial x} \right] + \frac{\partial}{\partial z} \left[ k(\psi) \frac{\partial \psi}{\partial z} \right] - \frac{\partial}{\partial z} k(\psi) + W \tag{5}
\]

Where

\[
\xi = \text{soil moisture capacity, } \xi = \frac{\delta \theta}{\delta \psi}
\]

\[
\psi = \text{pressure water head, cm}
\]

\[
K = \text{hydraulic conductivity, cm/hr}
\]

\[
W = \text{source/sink term}
\]

The solution of Eq.(5) requires the knowledge of the water retention and hydraulic conductivity curves. The Van Genuchten models will be used to describe those curves.

\[
\Theta = \frac{\theta - \theta_{r}}{\theta_{s} - \theta_{r}} = \left[ 1 + (\alpha \psi)^{n} \right]^{-m} \quad K(\Theta) = k_{s} \Theta^{0.5} \left[ 1 - \left( \Theta^{1/m} \right) \right]^{2} \tag{6}
\]

where

\[
\Theta = \text{normalized water content}
\]

\[
\theta_{r}, \theta_{s} = \text{residual and saturated volumetric water content, cm}^{3}/\text{cm}^{3}
\]

\[
\alpha, n, m = \text{fitting parameters within a given set of capillary pressure-saturation data.}
\]

\[
k_{s} = \text{saturated hydraulic conductivity, cm/hr.}
\]

A FORTRAN-77 computer program has been designed to solve the described mathematical model (Eq.3-6) [11]. The input parameters used in the simulation are illustrated in Table 4. Stagnant and recharge flow conditions prior to the simulation was applied to evaluate the effect of the initial flow conditions on the radionuclide transport behavior pulse and constant release scenarios. Figure 3 presents the simulated normalized concentration distribution profile for the Tc-99 under both flow conditions. The results shown in this figure could be attributed to the existence of a pressure head gradient through the unsaturated barrier due to the application of initial recharge condition, which will enhance the transport of Tc-99. By comparing the concentration at the ground water table with the MCL (Maximum Concentration Level) (5.3*10^{-5} mg/l), it was found that the concentration of Tc-99 for constant release scenario at the receptor point in a site under recharge conditions have exceed the MCL which mean that these site conditions will impose a potential risk to human health and the environment.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
<th>Unit</th>
<th>Parameter</th>
<th>Value</th>
<th>Unit</th>
</tr>
</thead>
<tbody>
<tr>
<td>Concentration of Tc-99 in recharge</td>
<td>0.0125</td>
<td>mg/l</td>
<td>0r</td>
<td>0.068</td>
<td>m³/m³</td>
</tr>
<tr>
<td>α</td>
<td>0.8</td>
<td>m¹</td>
<td>0s</td>
<td>0.38</td>
<td>m³/m³</td>
</tr>
<tr>
<td>N</td>
<td>1.9</td>
<td>Ks</td>
<td>0.048</td>
<td>m/day</td>
<td></td>
</tr>
<tr>
<td>Tortuosity constant, a</td>
<td>0.003</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Kd</td>
<td>0.007</td>
<td>ml/g</td>
<td>Do</td>
<td>4*10^{-6}</td>
<td>m²/day</td>
</tr>
<tr>
<td>α</td>
<td>2*10^{-4}</td>
<td>m²/day</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

323
4. Conclusion

This work represents the Egyptian radioactive waste management system with special reference to the disposal element including the site selection studies, engineering design, and performance assessment studies. The major emphasis of the present work is the evaluation of the performance of the disposal site under different scenarios for both short and long-lived radionuclides. The results indicated that loading the disposal facility with short-lived radionuclides, i.e. Cs-137, Sr-90, and Co-60, will not impose any harm to the human health and the environment. Where in sitting the long-lived radionuclides more attention should be given to determination of the inventory of these radionuclides.

REFERENCES


Treatment and management of low level radioactive waste at Institute for Technology of Radioactive and Rare Elements (ITRRE), Vietnam

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Vietnam

Abstract. The Institute for Technology of Radioactive and Rare Elements (ITRRE) has the task of treatment and management of all kinds of radioactive wastes in the North of Vietnam. Liquid wastes generated from different laboratories and hospitals have been collected and treated at the treatment and temporary storage facility in ITRRE. Solid wastes have been treated by the cementation technique after kept in the temporary storage facility. At present, 322 drums of solid wastes treated are stored safely in ITRRE.

1. Introduction

In the past decades, the applications of the nuclear technology in the field of industry, health service, geological survey, petrol exploitation and researching were rapid developed in Vietnam. Every year radioactive wastes are generated from laboratories, hospitals and industry.

ITRRE is a research establishment belonging to VAEC with the task of radioactive and rare ores treatment and processing, production of materials from Vietnamese mineral resources of radioactive, rare and non-ferrous elements.

2. Description of sources of radioactive wastes

Activities in research and development from laboratory to pilot scale have generated a rather large amount of radioactive wastes. Activities generating most radioactive wastes includes:

(a) Experimental treatment of uranium ores in bench scales - from several to few dozen tons of ore each time, for studying the possibilities of uranium recovery from different sources of domestic uranium deposits. The largest amount are waste ores after being processed for recovery of uranium (milling wastes) containing many radioactive elements such as radium; unrecovered uranium, etc. These are low level radioactive wastes but due to technological treatments the radio-nuclides in the ores become more mobile than before and can diffuse into the environment more easily. Then, it is necessary to store such kind of waste to avoid to make the environment contaminated with the mill tailings.

(b) Solid wastes of monazite concentrate after processing for rare-earths recovery in a pilot plant of monazite processing. The waste includes undissolved monazite concentrates and precipitates of thorium, radium and lead separated during the rare earth separation. This waste has a rather high activity, so it need to have suitable conditioning and careful management.

(c) Radioactive liquid wastes generated from research activities of ITRRE are usually of very low level, and mostly are natural radio-nuclides. Such wastes are collected and treated according to a predefined procedure.

Most of these wastes were produced during the operation of the Institute since 1980, mainly from activities of research, pilot production of uranium concentrate and monazite processing. Amount of 130 tons of low level radioactive wastes have been kept in temporary storage facility at Phung branch (it is 20 km far from Hanoi). In 2004, our Government has supported us to build a facility for treatment and temporary storage of radioactive wastes.
3. Treatment and management of radioactive wastes

3.1. Treatment of radioactive wastes in 2002

In the year 2002, ITRRE had treated in the first step the part of rather high level solid waste from monazite processing with the total amount of about 5 tons. Conditioning was conducted with cementation technique [1].

The treatment process includes: mix the waste with cement and additives according to the ratio: wastes: gravel: cement = 20: 60: 20. The mixture was mixed thoroughly; poured into casks and keep for solidification. Finally, the casks are moved into the temporary storage before they can be transported to the final repository. There are 50 drums of solid wastes treated. Radioactive dose rate of casks is $8 \mu$Sv/h at distance 1 m from surface of casks [1].

3.2. Treatment of liquid waste

Radioactive liquid wastes generated from research activities of ITRRE are usually of very low level, and mostly are natural radio-nuclides. such wastes are collected and processed according to a predefined procedure, which includes:

(i) Collection of radioactive liquid wastes in laboratories and transportation to the treatment section.

(ii) Precipitation of radio-nuclides in the solution, let the precipitates settle down.

(iii) Discard the solution into the sewerage system with suitable dilution. To recover the precipitates and transport to place of storage.

At the same time, some experiments have been conducted in the laboratory for treatment of radioactive wastes in the field of adsorption of uranium and thorium from waste water by zeolite NaA-CN93 made in Vietnam [2].

Exchange adsorption capacity of zeolite NaA-CN93 have been determined for uranium and thorium: 6.4mg U/g zeolite and 5.9 mg Th/g zeolite. Effluent after be processing by zeolite can be discarded to the environment [2].

3.3. Treatment of solid wastes in 2004

In the year 2004, amount of waste about of 80 tons in temporary vault, which needed to be processed and managed for safety. These wastes almost are mill tailing from uranium processing and sludge from liquid water treatment.

In general, radioactive activity of these wastes is low level. These wastes were separated into 3 types according to the physical characteristics: S1, S2 and S3.

S1- from sandstone after leaching of uranium ore
S2- waste from uranium ore processing
S3- sludge from liquid waste treatment

These samples have been analysed and results are presented in Table1 [3].

<table>
<thead>
<tr>
<th>N</th>
<th>Th (g/kg)</th>
<th>U(g/kg)</th>
<th>Activity (Bq/kg)</th>
<th>$\mu$Sv/h</th>
</tr>
</thead>
<tbody>
<tr>
<td>S1</td>
<td>0.001</td>
<td>6.76</td>
<td>83,552</td>
<td>4.4</td>
</tr>
<tr>
<td>S2</td>
<td>0.61</td>
<td>2.75</td>
<td>36,499</td>
<td>1.9</td>
</tr>
<tr>
<td>S3</td>
<td>0.03</td>
<td>20.7</td>
<td>255,893</td>
<td>13.5</td>
</tr>
</tbody>
</table>

The experiments for treatment of radioactive waste have been conducted in the laboratory for determination of suitable technological flow-sheet. Procedure have been applied for treatment of rad-wastes is cementation technique.
The parameters of flow-sheet: The suitable time of mixing is about 30 minutes. After that mixed mixture has been poured into drums and kept for solidification in 30 h [3].

We have determined the ratio (quantity) of components for 3 types of wastes as follow [3]:

S1: waste : cement : water = 1 : 0.3 : 0.2
S2: waste : cement : sand : water = 1 : 0.3 : 0.25 : 0.15
S3: waste : cement : sand : water = 1 : 0.5 : 0.5 : 0.15

The cement solidification for wastes have been done according to the flow-sheet above. The total of drums is 272. After 21 days we determined the press-intensity of concrete mass [3]:

Table 2. The press-intensity of concrete samples

<table>
<thead>
<tr>
<th>Sample</th>
<th>Press-intensity (Kgf/cm²)</th>
</tr>
</thead>
<tbody>
<tr>
<td>S1</td>
<td>63.8</td>
</tr>
<tr>
<td>S2</td>
<td>52.3</td>
</tr>
<tr>
<td>S3</td>
<td>23.5</td>
</tr>
</tbody>
</table>

The dose rate of different drums has been determined and is presented in Table 3 [3].

Table 3. Dose rate of different drums

<table>
<thead>
<tr>
<th>Samples</th>
<th>Dose rate before treatment (μSv/h)</th>
<th>Dose rate after treatment at distance of 1m (μSv/h)</th>
</tr>
</thead>
<tbody>
<tr>
<td>S1</td>
<td>23.8</td>
<td>4.05</td>
</tr>
<tr>
<td>S2</td>
<td>28.0</td>
<td>4.80</td>
</tr>
<tr>
<td>S3</td>
<td>39.5</td>
<td>5.50</td>
</tr>
</tbody>
</table>
Management of treated radioactive solid waste: drums were put in the temporary radioactive waste storage facility. Nowadays, in the temporary storage there are 322 drums. The dose rate is being determined periodically.

4. Conclusion

The treatment and management of radioactive wastes have been conducted safely at the Center of Radioactive Waste Management and Environment (ITRRE). Our temporary storage facility has responsibility for the management of all kinds of radioactive wastes in the North of Vietnam.

REFERENCES

Development of a low level radioactive waste repository in the Philippines

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Philippines

Abstract. This paper presents the initiatives and current status of project activities that have been implemented in the management of radioactive wastes in the Philippines. The paper deals primarily on siting, design, regulations, and safety assessment aspects that will support the development of a low level waste repository in the country. Preliminary results of site investigation identified 3 candidate sites having the most potential characteristics for the location of a repository. The proposed design concept involved a structure constructed below the ground above the water table. Safety assessment following international methodology is being conducted throughout the site selection process.

1. Introduction

Since the early 70’s, the Philippines has been engaged in the treatment, conditioning and storage of low and intermediate radioactive wastes in the Philippines. A great deal of work has been done to address the various issues associated with the management of radioactive wastes especially in recent years. These include drawing up policy, regulations and standards, planning and siting of a radioactive waste repository. This report presents the latest achievements in radioactive waste management in the Philippines.

2. National radioactive waste management policy

A national policy on Radioactive Waste Management serves as a national commitment to address the country’s radioactive wastes in a co-ordinated and cost effective manner. The scope of this policy relates to all radioactive wastes except operational radioactive liquid and gaseous effluents, which is permitted to be released to the environment routinely under the authority of the regulatory agency. Salient points include the following:

- protection of human health and the environment now and in the far future,
- generation of radioactive waste shall be kept to the minimum practicable,
- radioactive waste shall be managed within an appropriate national legal framework, including clear allocation of responsibilities and the provision for independent regulatory functions,
- the DENR in consultation with the PNRI shall be the responsible organizations over the final disposal of radioactive waste,
- the generators of radioactive waste or operators of radioactive waste disposal facilities, as the case may be, shall be responsible for the technical, financial and administrative management of such waste within the national regulatory framework and within any applicable co-operative governance arrangements.

3. Regulations and standards

The Philippine Nuclear Research Institute has been continuously formulating regulations, standards and criteria for the proper management of radioactive wastes. These regulations will be directed at protecting human health and the environment from potential releases of radioactive materials. Some of these
standards and regulations have been modified and updated to consider the current issues relating to the protection not only of humans but also that of the non-human biota:

- CPR Part 3. Standards for Protection Against Ionizing Radiation,
- CPR Part 23: Licensing Requirements for the Land Disposal of Radioactive Wastes,
- Standards on Facility Design,
- Preliminary Siting Factors and Associated Criteria.

4. Waste inventory

Initially, a database of all the wastes stored at the PNRI compound has been established to address the issues of waste treatment, transport, and interim storage. Work is now in progress to collate national inventory data so as to facilitate decisions on radioactive waste management strategy to support the development of operational and design requirements for a disposal facility. Consideration is being given to current arisings and to potential future arisings, allowing for alternative scenarios for the use of radioactive materials in the country. The volumes of solid, liquid and spent sealed sources received at PNRI according to records are summarized in Tables 1-4. Solid wastes were stored in the two trenches of the PNRI waste storage building. Almost all spent sources have been conditioned in 200 l drums.

Table 1. Current Inventory and Estimated Arisings Up to Year 2010
(Raw Waste Volumes)

<table>
<thead>
<tr>
<th>WASTE PRODUCER</th>
<th>TYPICAL WASTE</th>
<th>ESTIMATED CURRENT VOLUME (2004), m³</th>
<th>*ESTIMATED FUTURE WASTE ARISINGS (2010), m³</th>
</tr>
</thead>
<tbody>
<tr>
<td>PNRI</td>
<td></td>
<td>Solid</td>
<td>Liquid</td>
</tr>
<tr>
<td>PNRI Licensees</td>
<td>Glasswares, paper, clothing, lab equipment Research reactor liquid wastes IX resins</td>
<td>105</td>
<td>24.7</td>
</tr>
<tr>
<td></td>
<td>Medical sources, hospital wastes, gauges, smoke detectors, glasswares, paper, clothing, used sources from medical, industrial and research</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PNRI Licensees</td>
<td>Disused sources</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

NOTE: 1. *Based on an annual arising of 7 m³/a solid waste and 1 m³/a for liquid waste
2. Isotopic Content: $^3$H, $^{32}$P, $^{131}$I, $^{125}$I, $^{90}$Sr, $^{89}$Sr, $^{137}$Cs, $^{60}$Co, $^{59}$Co, $^{14}$C
**Scenario 1: future arisings remain at present levels**

Table 2. Conditioned Drum Wastes
(as of 2001)

<table>
<thead>
<tr>
<th>WASTE PRODUCER</th>
<th>TYPICAL WASTE</th>
<th>NO. OF DRUMS</th>
</tr>
</thead>
<tbody>
<tr>
<td>PNRI and PNRI Licensees</td>
<td>Glasswares, paper, clothing, lab equipment</td>
<td>455</td>
</tr>
<tr>
<td></td>
<td>Research reactor liquid wastes</td>
<td></td>
</tr>
<tr>
<td></td>
<td>IX resins</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Medical sources, hospital wastes, gauges, smoke detectors, glasswares, paper,</td>
<td></td>
</tr>
<tr>
<td></td>
<td>clothing, used sources from medical, industrial and research</td>
<td></td>
</tr>
<tr>
<td>PNRI Licensees</td>
<td>Disused sources</td>
<td>150</td>
</tr>
</tbody>
</table>

**Scenario 2: the PRR-1 becomes operational plus 10% increase in arisings**

Table 3. Conditioned Drum Wastes
(As of 2001)

<table>
<thead>
<tr>
<th>WASTE PRODUCER</th>
<th>TYPICAL WASTE</th>
<th>No. of Conditioned Waste Drums</th>
</tr>
</thead>
<tbody>
<tr>
<td>PNRI</td>
<td>Glasswares, paper, clothing, lab equipment</td>
<td>887</td>
</tr>
<tr>
<td>PNRI Licensees</td>
<td>Research reactor liquid wastes</td>
<td></td>
</tr>
<tr>
<td></td>
<td>IX resins</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Medical sources, hospital wastes, gauges, smoke detectors, glasswares, paper,</td>
<td></td>
</tr>
<tr>
<td></td>
<td>clothing, used sources from medical, industrial and research</td>
<td></td>
</tr>
<tr>
<td>PNRI Licensees</td>
<td>Disused sources</td>
<td>192</td>
</tr>
</tbody>
</table>
Scenario 3: the PRR-1 was decommissioned

Table 4. Estimated future waste arisings generated by the decommissioning of the Philippine Research Reactor

<table>
<thead>
<tr>
<th>WASTE PRODUCER</th>
<th>TYPICAL WASTE</th>
<th>*ESTIMATED FUTURE WASTE ARISINGS, m³</th>
</tr>
</thead>
<tbody>
<tr>
<td>PNRI</td>
<td>Core support structures, concrete, tools, residues and compactible wastes</td>
<td>75</td>
</tr>
</tbody>
</table>

*Note. Initial estimate given by an IAEA Expert. Final projection will be determined when more information including the choice of decommissioning technology will be established.

5. Site selection activities

The Philippines is currently selecting a site for a national near surface repository for the disposal of low to short lived intermediate radioactive wastes. The overall siting process was divided into 2 phases. Phase I involves site screening procedure which mainly deals with desk compilation of existing information and data sourced from different government organizations. The initial nationwide selection was based on physical features related to the site’s subsurface hydrology, surface hydrology, geology and soils, topography, meteorology and land use. Figure 1 shows the 35 sites selected for possible location of the proposed disposal facility.

The Phase II of the project was undertaken between the years 2003-2004. Site surveys and validation were conducted to shortlist the number of sites that will be subject to a more detailed sub-surface investigation. Based on the results of preliminary field investigation and available technical data, the final three (3) candidate sites depicted in Fig. 1 were identified for detailed site assessment study. The objective of the site assessment is to establish the geological, geomorphologic and hydro-geological characteristics of each of the 3 sites and rank them in the order of suitability for subsequent detailed sub-surface investigation. In consideration of the vast resources needed to perform detailed sub-surface investigation, the assessment has led to the ranking of the relative suitability of each of the 3 sites. The technical working committee agreed that a single site would be identified thereby allowing focused work activities on this single particular location.

6. Safety assessment

In the second half of 2005 it is anticipated that PNRI will be drawing together all the information on current waste inventory, anticipated future waste arisings, proposed site characteristics, facility design options and regulatory development and the regulator/operator/waste producer interaction process.

The assessment methods are expected to be used to test and confirm the effectiveness of proposed waste management options to meet internationally recognized safety requirements. Components of the options include:

- identification of the wastes which can be disposed of in a near surface facility;
- identification of suitable waste treatment and packaging options;
- identification of the advantages and disadvantages of different sites, operationally and post-closure;
- identification of the advantages and disadvantages of potential design options;
• assumptions for long-term site management, including institutional control;
• identification of regulatory requirements for each stage of repository design, construction, operation, and closure; and
• operational ‘Conditions for Acceptance’ for wastes proposed for disposal; to be developed by the operator as information to be provided to waste producers who wish to use the disposal facility.

FIG. 1: Possible Sites for Near Surface Waste Disposal

All this work can then place the Sub-committee on Radioactive Waste Management and the PNRI in a good position to decide on the next steps in application of radioactive waste management strategy beyond 2005.
Initially, the proposed disposal concept is to build an engineered structure above the water table. The proposed repository design involves an area of 1.2 x 1.2 sq. km with a total capacity of 10,000 cubic meters good for about 60 years operation. This concept is expected to be evolving since more information such as the waste inventory, packaging and site characteristics become available. Safety assessment of the
disposal concept will be conducted with the use of appropriate mathematical models and having a time frame extending into hundreds of years. The overall purpose is to determine the feasibility in terms of meeting post-closure radiological protection objectives, of developing a low-level waste disposal facility. Further objectives include determination of effects on post-closure radiological impacts of alternative siting options (waste treatment and containment options, and alternative disposal facility engineering options. From the above, the key uncertainties and further information needs should be identified though sensitivity calculations and alternative assessment assumptions.

Following international methods, Table 5 shows the adopted list of important parameters identified for an undisturbed site. These pathways assume that the containment structure and soil cover remain intact and perform as designed to provide adequate shielding and minimize infiltration. The pathways for a disturbed site shown in Table 6 will occur after the occurrence of an event that disrupts the integrity of the facility and exposes the wastes at the surface. Figure 3 shows the different models and computer codes to assess the two most important pathways, i.e. groundwater and surface water pathways, for the performance of a low level waste facility.

**Table 4. Proposed Repository Design**

<table>
<thead>
<tr>
<th>AREA</th>
<th>1.2 km x 1.2 km</th>
</tr>
</thead>
<tbody>
<tr>
<td>ACTUAL DISPOSAL AREA</td>
<td>100 meter x 100 meter</td>
</tr>
<tr>
<td>NUMBER OF TRENCHES</td>
<td>TWO (2) trenches less than 20 meters deep</td>
</tr>
<tr>
<td>CAPACITY</td>
<td>10,000 cubic meter for about 60 years operation</td>
</tr>
<tr>
<td>COVER SYSTEM</td>
<td>Multi layer cover system made from natural materials; engineered cover system</td>
</tr>
<tr>
<td>ACTIVE AREA</td>
<td>From 5 meters to 10 meters below the surface</td>
</tr>
</tbody>
</table>

**Table 5. Generic Pathways for Undisturbed Performance of Low Level Waste Disposal**

<table>
<thead>
<tr>
<th>SOURCE</th>
<th>Groundwater</th>
<th>MAN</th>
<th>MAN</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Soil</td>
<td>Plants</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Land Plants</td>
<td>Land</td>
<td>Animals</td>
</tr>
<tr>
<td></td>
<td>Land Animals</td>
<td>Aquatic</td>
<td>Animals</td>
</tr>
<tr>
<td></td>
<td>Surface Water</td>
<td>Soil</td>
<td>Land</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Plants</td>
<td>Animal</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Land</td>
<td></td>
</tr>
</tbody>
</table>

Source: Shipers and Harlan, 1989
FIG. 3: Models and codes
Table 6: Generic Pathways for Disturbed Performance of Low Level Waste Disposal

<table>
<thead>
<tr>
<th>SOURCE</th>
<th>Man</th>
<th>Air</th>
<th>Man</th>
<th>Man</th>
<th>Man</th>
<th>Man</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Soil</td>
<td>Man</td>
<td>Soil</td>
<td>Land plants</td>
<td>Man</td>
<td>Soil</td>
</tr>
<tr>
<td></td>
<td>Land plants</td>
<td>Soil</td>
<td>Land plants</td>
<td>Man</td>
<td>Man</td>
<td>Soil</td>
</tr>
<tr>
<td></td>
<td>Man</td>
<td>Man</td>
<td>Man</td>
<td>Man</td>
<td>Man</td>
<td>Man</td>
</tr>
</tbody>
</table>

Source: Shipers and Harlan, 1989

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Dose and risk assessment of NORM contaminated waste release from trench disposal and salt caverns

M. Abdel Geleel, A.B.A. Ramadan, A.A. Tawfik
National Centre for Nuclear Safety and Radiation Control,
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Abstract. Oil and gas extraction and processing operations sometimes accumulate naturally occurring radioactive materials (NORM) at concentrations above normal in by-product waste streams. There are number of industries generating NORM contaminated waste in Egypt. Results from NORM surveys indicate that radionuclide concentrations can be quite variable, ranging from undetectable to extremely high levels. The petroleum industry adopted methods for managing NORM-contaminated wastes that are more restrictive than past practices and are likely to provide greater isolation of the radioactivity. A trench was used as a disposal facility for NORM contaminated waste at one site of the petroleum industry in Egypt. Some countries encourage the disposal of NORM waste onto salt caverns. The aim of this work is to calculate the risk and dose assessment received from the trench disposal facility and also from salt caverns after direct closure and after post closure (1000 year). The RESRAD computer code with two different scenarios was used for this purpose. The results indicate that the total effective dose (TED) received after direct closure of trench and salt cavern disposal facilities was 7.7E-4 and 4E-9 mSv/y respectively, while after 1000 years it was 3.4E-4 and 2.6E-8 mSv/y. The health cancer risk from both facilities after direct closure was 3.3E-8 and 3.1E-11, while after 1000 years post closure it was 6E-8 and 1.2E-12 respectively. Results of this assessment can help to examine policy issues concerning different options and regulation of NORM contaminated wastes generated by the petroleum industry.

1. Introduction

NORM, or naturally occurring radioactive material, is found almost everywhere. It is found in the air and in soil, and even in radioactive potassium in our own bodies. It is found in public water supplies and foods such as brazil nuts, cereal, and peanut butter.

Oil and gas production, and processing operations, sometimes accumulate NORM at elevated concentrations in by-product waste streams. The sources of most of the radioactivity are isotopes of uranium-238 (U-238) and thorium-232 (Th-232), which are naturally present in subsurface formations from which oil and gas are produced. The primary radionuclides of concern in NORM wastes are radium-226 (Ra-226) of the U-238 decay series and radium-228 (Ra-228) of the Th-232 decay series. Other radionuclides of concern include radionuclides that form from the decay of Ra-226 and Ra-228, such as radon-222 (Rn-222). The production waste streams most likely to be contaminated by elevated radium concentrations include produced water, scale, and sludge [1]. Radium, which is slightly soluble, can be mobilized in the liquid phases of a formation and transported to the surface in the produced water stream. Dissolved radium either remains in solution in the produced water or precipitates out in scales or sludge. Conditions that appear to affect radium solubility and precipitation include water chemistry (primarily salinity), temperature, and pressure. NORM contamination of scale and sludge can occur when dissolved radium co-precipitates with other alkaline earth elements such as barium, strontium, or calcium [2].

NORM waste is physically and chemically similar to non-hazardous oil field waste (NOW). Its primary difference from NOW is the presence of radionuclides in NORM waste. The presence of radionuclides may require additional safety precautions when handling the NORM waste, but the actual disposal process would be no different from that for NOW [3]. The petroleum industry adopted methods for managing and disposing of NORM-contaminated wastes that are more restrictive than past practices and are likely to provide greater isolation of the radioactivity. Simultaneously, many countries have promulgated NORM regulations that establish new, more restrictive standards for the...
management and disposal of NORM wastes. These actions have served to limit the number of available disposal options for NORM wastes, thereby increasing waste management costs. The largest volume oil and gas waste stream that contains NORM is produced water. Except at offshore platforms, which discharge produced water to the ocean, nearly all produced water is injected into the subsurface through injection wells. At this time, the radium content of produced water going to injection wells is not regulated. Consequently, radium that stays in solution in the produced water stream does not present a significant waste management problem from a regulatory perspective and is not considered further in this study. Some operators dispose of NORM wastes at their own sites although, most use off-site commercial disposal facilities. Pipes and casing with NORM contamination may be recycled as scrap steel if NORM levels are below background concentrations. In the past, NORM was commercially managed by surface treatment, through which NORM was blended with non-radioactive materials to reduce the NORM activity below action levels and to spread on the land. Today, the primary method used for disposal of NORM wastes is underground injection. Smaller quantities of NORM waste are disposed of at licensed radioactive waste landfills, encapsulated in the casing of a well being abandoned, or managed on lease sites through land spreading [4].

The safe handling of these NORM in Egypt implies identification of the responsibilities of both the producers of the NORM and the Central Radioactive Management Authority. In Egypt, this authority is the Hot Laboratory and Waste Management Centre (HLWMC) [5]. Dealing with these wastes requires developing both the required technologies and the relevant regulations to determine the responsibilities and identify the safety requirements for the handling of such wastes. The responsibilities of the producer include waste collection, packaging of category 1 and interim storage of category 2. The responsibilities of the HLWMC include transportation and long term storage of category 1. In this study, the risk and dose assessment received from trench disposal facility and also from salt caverns after direct closure and after post closure (1000 year) was calculated. RESRAD computer code with two different scenarios was used for this purpose.

2. NORM management practices

NORM was not recognized as a waste management issue, however, until the mid-1980s, when the industry and regulators realized that NORM occurrence was more widespread than originally thought and that activity levels could be high. The petroleum industry adopted methods for managing and disposing of NORM-contaminated wastes that are more restrictive than past practices and are likely to provide greater isolation of the radioactivity.

Underground injection

NORM-contaminated scales, sludges, and other solid wastes have also been disposed of through underground injection wells [6]. Report on a NORM waste injection project in the North Slope Alaska oil field developed by two major producing companies. Approximately 100 tons of NORM solids were cleaned from 3,000 oil production pipes and casing. The resulting solids were processed to a particle size of less than 80 micrometers (Fm), slurried with 10,000 bbl of water, and then injected into a Class II injection well [6]. Other disposal aspects will come to an operator's site and process NORM wastes so that they can be injected through the operator's own injection well. The process consists of grinding and milling the waste to a small particle size, slurrying the waste to facilitate pumping, and injecting to formations at fracture pressure [7].

Landfill disposal

Burial in landfill is another off-site commercial NORM waste disposal option.

Encapsulation and downhole disposal

Under the encapsulation and downhole disposal option, an operator encapsulates NORM waste either inside a section of pipe that is then sealed on both ends and lowered into a wellbore or directly in the wellbore. A plug is placed on top of the waste-containing zone. Two encapsulation projects conducted in the offshore Gulf of Mexico [8]. In the first project, NORM waste was placed into eight joints of casing as the pipe was being lowered into the hole. In the second project, 31 drums of NORM waste were placed into 21 joints of casing on shore and sealed on both ends. The sealed joints were transported offshore and lowered into the well bore. In both project, cement plugs were placed on top
of the waste containing joints. Encapsulation works well for NORM waste disposal, but each well can handle only a relatively small volume of waste. Because of this restriction, the process is not widely used.

**Land spreading**

The principle behind land spreading is to mix NORM wastes having an activity concentration higher than the action level with clean soil so that the resulting blend has an activity concentration lower than the action level. Some producers utilize land spreading on their lease site to blend patches of high-activity NORM soils with low activity NORM soils. However, the present use of land spreading for disposal of NORM waste is limited.

**Trench disposal**

It is another option adopted in Egypt and applied at one petroleum industrial.

**Disposal cavern operation**

Initially, the caverns would be filled with brine. Wastes would then be introduced as a slurry of waste and a fluid carrier (brine or fresh water). Three scenarios are possible for introducing the waste material: (1) the waste can be pumped down tubing to the bottom of the cavern and the displaced brine can be withdrawn through an annulus; (2) the waste can be pumped down an annulus and the displaced brine can be withdrawn through the tubing; and (3) the waste can be injected through one well and the brine withdrawn from another well. As the slurry is injected, the cavern acts as an oil/water/solids separator. The heavier solids sink to the bottom of the cavern and form a pile. Any free oils and hydrocarbons float to the top of the cavern because they are less dense than water. An organic blanket could be injected into the cavern to prevent additional leaching of the cavern's roof by water that is not fully saturated with salt. Clays in the slurry and dissolved chemical constituents from the waste can mix with the brine, forming a suspension above a brine/waste interface. Clean brine displaced by the incoming slurry would be removed from the cavern and either sold as a product or disposed of in an injection well. Early in the life of the disposal cavern, clean brine is withdrawn from hundreds of feet above the surface of the waste pile or interface. As the cavern fills, the brine becomes dirtier (i.e., it will have a higher clay, oil, and dissolved waste constituent content). This dirty brine can produce operational difficulties (e.g., clogging of pumps) and additional expenses. The cavern is considered to be “full” of waste when return of disposed material with the displaced fluid becomes a problem. When the cavern is full, the operator seals the cavern.

**Post-closure cavern behaviour**

Once the cavern had been filled with waste, the cavern would be sealed and the borehole plugged with cement. Plugs would be placed in the well bore above and below water-bearing intervals to isolate these intervals permanently. During a transient period of several years after closure of a cavern filled with brine, pressure can exceed the litho static value (pressure in surrounding salt) because of thermal expansion of the brine. The amount of over pressurization is a function of cavern size. Similarly, cavern pressure can exceed the litho static value after a longer time period when, due to salt creep, brine pressure will balance average lithostatic pressure, resulting in a slight excess of brine pressure at the top of the cavern. This over pressurization occurs because litho static pressure increases linearly with depth, whereas brine pressure is constant within the cavern.

3. **Exposure assessment**

Migration of radionuclides may occur when water comes into contact with NORM contaminated waste and carries the radionuclides into the surrounding soil. The radionuclides are likely to migrate more rapidly when coarse-grained deposits, like sand and gravel, exist in the surrounding soil and reaches to the groundwater. Once contaminated fluids leave the disposal facility (trench or caverns) they are expected to migrate laterally through different formations and aquifers. During the time the fluids travel from the point of release to the receptor site, various physical, chemical, biological, and radiological processes occur that reduce the concentration of the contaminants. In this study, exposed individuals are expected to be those drinking groundwater contaminated by releases of NORM constituents from disposal facilities containing NORM wastes. The exposure pathway would consist of release from the facility, transport through groundwater, and human exposure through ingestion of the
contaminated groundwater. This study describes the scenarios and mechanisms that could lead to human exposure to NORM constituents and estimates radiological doses and human health risk to a potential receptor. Once the facility was full of waste, it would be covered and abandoned. At the time of sealing, the facility would be mostly filled with solids and semi solids that were not fully compacted.

3.1. Detection and measurements of NORM

The detection and measurement of NORM may be subdivided into four key procedures:

a) External monitoring using hand-held radiation detectors for penetrating beta and gamma radiation to identify locations of NORM and to identify any possible hazards to workers in the vicinity thereof. Monitoring can be carried out inside equipment during maintenance and down-time.

b) Measurements using radiation spectrometers on site to identify the specific type of radiation emitted. Gamma radiation is emitted by radionuclides at specific energies which permit identification. Quite rugged and portable devices are available for use by trained personnel. Special types of detectors are available for the measurement of Radon concentrations.

c) Collection of sample material (e.g. water, scales and sludges) and assay by radio-chemical laboratories to determine the type of NORM and concentration in samples.

d) Measurement of the radiation exposure to workers integrated over long intervals by the use of film badges, or thermo-luminescent devices (TLDs) worn by workers. Usually such measurements will be carried out in collaboration with regulatory agencies or authorized Health-Physics professionals.

3.2. Collection of materials and methods

Five samples weighing about 1 kg each were collected from the NORM waste that will be disposed into the trench or salt caverns. The waste samples contains the sludge and scale formed in the production equipment, during oil and gas extraction, and those removed during the periodical maintenance. The samples pulverized to fine powder (200 mesh) and then dried over night at 110 °C. Known amount of the samples (50 gm) were packed in a special counting container and carefully sealed for 8 wk to reach secular equilibrium between 232-Th and 238-U progeny. The activity of 214Bi, 214Pb, and other progenies of 228Ra in equilibrium with their parents were assumed to represent the 238-U activity. The activity of 228Ac and other progenies of 232-Th were assumed to represent 232-Th activity. Sludge samples were stirred manually by a glass rod and filled in a plastic container for counting. Radiometric analyses of different samples were carried out using a high resolution gamma ray spectrometric system. The system was comprised of high purity germanium detector (HPGe) with 81 cm3 sensitive volume. The detector has an energy resolution of 2.2 keV FWHM for the 1,332.5 keV of gamma energy of 60Co. To reduce environmental gamma background radiation, the detector was shielded with lead bricks of thickness 5 to 10 cm. A lining of 2-mm-thick copper followed by 2-mm-thick aluminum was made to absorb the x rays from lead and copper.

4. Regulations concerning NORM

According to the national laws and regulations it is not allowed to dispose or release any radioactive waste or contaminated materials without authorization from the Egyptian Atomic Energy Authority.

Regulations concerning NORM have generally been derived form regulations which apply to the handling, transport and disposal of radioactive material.

<table>
<thead>
<tr>
<th>Table 5: Recommended dose limits(13)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Effective Dose</strong></td>
</tr>
<tr>
<td></td>
</tr>
</tbody>
</table>

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5. Result and discussions

5.1. Activity level in scale NORM waste

From above experiment the following radionuclides from the $^{238}$U series having half-lives exceeding one day was identified: $^{234}$Th, $^{234}$U, $^{230}$Th, $^{226}$Ra, $^{222}$Rn, $^{210}$Pb, $^{210}$Bi, $^{210}$Po. From the $^{232}$Th series we have: $^{228}$Ra, $^{228}$Th, $^{224}$Ra. The Highest level of radium isotopes ($^{226}$Ra, $^{228}$Ra, $^{224}$Ra) was observed from hard scale samples. Table 2 shows some of the analyses results of radium activity and it was ranged between 153 and 34 Bq/g.

<table>
<thead>
<tr>
<th>Sample no.</th>
<th>$^{226}$Ra (Bq/g)</th>
<th>$^{228}$Ra (Bq/g)</th>
<th>$^{224}$Ra (Bq/g)</th>
<th>$^{40}$K (Bq/g)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>71 ± 3</td>
<td>52 ± 4</td>
<td>34 ± 3</td>
<td>7 ± 1</td>
</tr>
<tr>
<td>2</td>
<td>82 ± 4</td>
<td>63 ± 3</td>
<td>49 ± 1</td>
<td>8.2 ± 1</td>
</tr>
<tr>
<td>3</td>
<td>153 ± 4</td>
<td>119 ± 1</td>
<td>72 ± 3</td>
<td>13 ± 1</td>
</tr>
<tr>
<td>4</td>
<td>138 ± 3</td>
<td>98 ± 5</td>
<td>64 ± 1</td>
<td>6.2 ± 1</td>
</tr>
<tr>
<td>5</td>
<td>106 ± 3</td>
<td>83 ± 3</td>
<td>55 ± 2</td>
<td>6.7 ± 1</td>
</tr>
</tbody>
</table>

5.2. Activity level in sludge NORM waste

Sludge containing NORM are produced from the cleaning of oil separator, storage tanks and other surface equipment. These wastes contain less activity than the hard scale and that is shown in table 3.

<table>
<thead>
<tr>
<th>Sludge Sample no.</th>
<th>NORM Concentration (Bq/g)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$^{226}$Ra</td>
</tr>
<tr>
<td>1</td>
<td>35 ± 1</td>
</tr>
<tr>
<td>2</td>
<td>72 ± 2</td>
</tr>
<tr>
<td>3</td>
<td>121 ± 3</td>
</tr>
<tr>
<td>4</td>
<td>113 ± 3</td>
</tr>
<tr>
<td>5</td>
<td>86 ± 3</td>
</tr>
</tbody>
</table>
5.3.  **RESRAD computer code**

The RESRAD computer code version 6.21 was developed under the joint sponsorship of the U.S. Department of Energy and the U.S. Nuclear Regulatory Commission for site-specific dose assessment of residual radioactivity.

Assumptions are as follows: NORM contaminated radionuclides were considered as potential A 70 Kg weight man drinks 2 litres of ground water from a well located 5000 m distance a way from the disposal site. The assumptions and the parameters used in this analysis depend on:

1- disposal site characterization (trench and salt caverns),
2- types of NORM waste arising,
3- design of the disposal trench and salt caverns,
4- the hydrology and geohydrology of the site,
5- the worst case; there is no unsaturated zone,
6- the cover and barrier is completely failed (no barrier exist).

The input data and the assumptions for trench disposal facility are:

- Porosity : 0.4
- Precipitation : 1 m/y
- Irrigation : 0.2 m/y
- Density : 1.5 g/cm³
- Hydraulic conductivity : 100 m/y
- Hydraulic gradient : 0.02.

Fig.1 shows a relation between the total effective dose (mSv) with the time (years) that received as a result of direct exposure. The maximum dose received from direct external exposure to all radionuclides is 8E-04 this may be due to the presence of $^{226}$Ra.

![Fig. 1: Total effective dose received from external exposure to all radionuclides and all pathways](image)

The excess cancer risks for all radionuclides and all pathways reached 1.25 E-07 and this is shown in Fig.2. Fig.3 shows a relation between a human risk against time and this figure indicated that trench disposal of NORM waste poses a very low human health risk. It is clear from the radioactive dose assessment results that the total annual exposure to the whole body is less than the limit of 0.25 mSv (10 CFR 61). This means that trench disposal is safe to dispose the NORM waste.
FIG. 2: Excess cancer risk from external exposure nuclides and all exposure pathways

**Fig. 3**: Dose/Source ratio for Ra-226 (external and all pathways)

**FIG. 4**: Dose/source ratio for Ra-228 (external and all pathways)

**FIG. 5**: Dose/Source ratio for K-40 (external and all pathways)
6. Conclusion

This study provides evidence that caverns disposal of NORM contaminated waste that produces from oil and gas production poses a very low human health risk and is most likely technically feasible compared with trench disposal facilities. The RESRAD computer code is easily used to help in preparing safety and risk assessment for both types of the disposal facilities.

REFERENCES


Current status of the regulatory system on radioactive waste management in the Russian Federation

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Abstract. The current status of the regulatory system on radioactive waste management in the Russian Federation is presented in the paper. The general principles of the regulatory system development and its correspondence with the international practice and the IAEA recommendations are discussed. The open items and future plans on improvement the regulatory system and its harmonization with the international practice are outlined.

1. Introduction

The system of regulatory documents on radioactive waste management as well as the legislative system of the Russian Federation in general is at the development stage. Some legal documents were developed and put into force, others are at the stage of development or approval, but some documents have to be developed in the future. Concentration of activity on regulations development within the state regulatory bodies (the Federal Environmental, Industrial and Nuclear Supervision Service of Russia, the Ministry of Public Health and the Ministry of Emergency Situation) as well as distribution of powers between regulatory bodies allows to harmonize and optimize the system of regulatory documents in the field of radioactive waste management.

2. Regulatory infrastructure

The hierarchy of the legal documents and regulations in the Russian Federation is, in general, similar to the structures adopted in other developed countries. The hierarchical scheme of the legal and regulatory documents in the field of radioactive waste management includes:

- the Constitution of the Russian Federation,
- the Federal Laws and International Agreements,
- the legislative acts of the President and Government of the Russian Federation,
- the federal regulation in the field of nuclear energy use,
- the guiding documents of the state safety regulatory authorities,
- the standards and industry regulations.

The primary Federal Laws defining the legislative basis for radioactive waste management are the following - “On the Use of Atomic Energy” No 170-FZ of November 21, 1995, “On Radiation Safety of Population” No 3-FZ of January 09, 1996 and “On the Environmental Protection” No 7-FZ of January 10, 2002 and some other Federal Laws. At present the main problem of the establishment of legislation basis for the state system of radioactive waste management is the lack of the Federal Law “On Radioactive Waste Management”, which shall set up the general principles for radioactive waste management at the law level as well as a clear allocation of powers and responsibilities of the state regulatory and control bodies (both, federal and local) and organizations involved in activities at the different steps of radioactive waste management. This Federal Law shall also establish the financial sources for radioactive waste management including those for the final stage – waste disposal, especially during the post-closure phase.

Based on the systematic approach "The Concept on Establishment of the System of Regulatory Documents on Safety Ensuring on Radioactive Waste Management” [1] was developed in the Russian Federation. It was agreed upon by the Ministry of Atomic Energy and the State Environmental Committee of Russia, and was approved and enacted by the Decree of the Gosatomnadzor of Russia No 8 of November 5, 1997 and by the Decree of the Health Ministry of Russia No 2 of January 15,
This concept stipulates the development of the following categories of regulatory documents within the regulatory system:

- “Safety Fundamentals” - presents basic safety objectives, fundamental safety principles and requirements,
- “Safety Requirements” - presents the requirements which should be followed up to ensure safety of specified activities,
- “Safety Guides” - presents recommendations on methods for compliance with the safety requirements.

The structure of legal and regulatory documents on radioactive waste management includes documents, enforced by the Federal Environmental, Industrial and Nuclear Supervision Service of Russia (Rostechnadzor) and related to regulation of technical and administrative aspects of safety ensuring, and also health rules and standards, enforced by the Ministry of Public Health and establishing sanitary-hygienic requirements and standards prescribing limits of permissible exposure to personnel and members of the public (for instance, OSPORB-99 [2] and SPORO-2002 [3]).

### 3. Safety requirements

A number of regulatory documents were developed by the Rostechnadzor in 2004 year including NP-058-04 “Safety of Radioactive Waste Management. General Provisions” [4]. This regulation establishes the general objectives and principles of safety ensuring in radioactive waste management, as well as general safety requirements. This regulation was developed taking into account provisions of the legal acts of the Russian Federation, the Joint Convention on the safety of spent fuel management and on the safety of radioactive waste management [5] and the IAEA recommendations [6]. According to NP-058-04 the following principles shall be met in radioactive waste management:

- provide for effective protection of personnel and population,
- provide for effective protection of environment,
- take into account interdependencies among the different steps in radioactive waste management,
- avoid actions that impose reasonably predictable impact on future generations greater than those permitted for the current generation (protection of a future generations),
- avoid imposing undue burdens on future generations,
- ensure that the generation of radioactive waste is kept to the minimum practicable,
- prevent the accidents with radiological consequences and mitigation of their consequences in case of its occurrence.

The regulatory document NP-055-04 “Radioactive Waste Disposal. Principles, Criteria and Basic Safety Requirements” [7] enforced by the Rostechnadzor in 2005 establishes the principles, criteria and general requirements to safety ensuring for near surface disposal of radioactive waste, deep geological disposal as well as liquid waste disposal (limited disposal of liquid waste into the deep geological layers is allowed only for three operating organizations within existing mining leases). The objective of safety ensuring for radioactive waste disposal is the reliable isolation of waste that will provide for radiological protection of human and the environment for the whole period of waste potential hazard. According to NP-055-04 the following principles shall be met in radioactive waste disposal:

- keep the radiological impact as low as practically achievable taking into account economical and social factors (optimization principle),
- ensure the long term safety of radioactive waste repository by the system of barriers (multiple barriers principle),
- protection of a future generations,
- avoid imposing undue burdens on future generations.
4. Classification of radioactive waste

According to the specific activity, the liquid and solid radioactive waste are classified into the following three groups (OSPORB-99 [2]): Low Level Waste (LLW), Intermediate Level Waste (ILW) and High Level Waste (HLW).

<table>
<thead>
<tr>
<th>Group</th>
<th>Specific activity, Bq/g</th>
<th>( \beta )-emitting</th>
<th>( \alpha )-emitting</th>
<th>transuranic</th>
</tr>
</thead>
<tbody>
<tr>
<td>LLW</td>
<td>below ( 10^3 )</td>
<td>below ( 10^2 )</td>
<td>below ( 10^1 )</td>
<td></td>
</tr>
<tr>
<td>ILW</td>
<td>( 10^3 - 10^7 )</td>
<td>( 10^2 - 10^6 )</td>
<td>( 10^1 - 10^5 )</td>
<td></td>
</tr>
<tr>
<td>HLW</td>
<td>above ( 10^7 )</td>
<td>above ( 10^6 )</td>
<td>above ( 10^5 )</td>
<td></td>
</tr>
</tbody>
</table>

Besides that, the regulatory document NP-055-04 [7] set up the maximum concentration of some radionuclides in radioactive waste packages authorized for near surface disposal (including a long-lived transuranic radionuclides – not more than 370 Bq/g in an overall average per waste package and 3700 Bq/g in individual waste packages).

At the same time it is necessary to note that present classification of radioactive waste requires the further development, taking into consideration the final management stage – waste disposal, and its harmonization in accordance with requirements of the Joint Convention (Article 11) [5], the IAEA recommendations [8] and with the sound international practice. This can be achieved by separation of the special subcategory of the intermediate long-lived waste with concentration of long-lived radionuclides greater than 370 Bq/g, which is the subject to deep geological disposal.

5. Clearance limits

It is established by OSPORB-99 that any restrictions in use of solid materials, raw material and products should not be imposed, if the specific activity of radionuclides containing in these items are less than 0,3 Bq/g and gamma dose rate at 0,1 m distance from its surface is less than 0,2 \( \mu \)Gy/h (over background level). The permissible specific activity of 14 main radionuclides for unconditional use of metal after licensed melting are also given in OSPORB-99.

For conditional release from regulatory control of raw material, materials and products (for the purposes of its subsequent use on I-III category of nuclear facilities) it is established the following levels of permissible specific activity:

(i) from 0,3 to 100 Bq/g for beta-emitting nuclides;
(ii) from 0,3 to 10 Bq/g for alpha-emitting nuclides;
(iii) from 0,3 to 1 Bq/g for transuranic elements;
(iv) from 0,2 to 1,0 \( \mu \)Gy/h of gamma dose rate at 0,1 m distance from its surface (over background level).

In case when the use of these materials are impossible or unreasonable it shall be disposed at an industrial dumps. The permissible amount of very low level radioactive waste intended for disposal at an industrial dumps shall be agreed with the regulatory authorities.

REFERENCES


Abstract. This paper presents the current status of radioactive waste management in general in Kenya. Sources, management and classification of radioactive waste are described.

1. Introduction

All uses of radionuclides in agriculture, research, medicine and industry, as well as activities associated with nuclear power production, generate waste much in the same was as do other human activities. These wastes contain radionuclides that emit ionizing radiation and are referred to as radioactive wastes. The International Atomic Energy Agency (IAEA) defines radioactive waste as “any material that contains or is contaminated with radionuclides at concentration or radioactivity levels greater than the ‘exempt quantities’ established by the competent regulatory authorities and for which no use is foreseen”. The ‘exempt quantities’ of wastes are those containing such low levels of radioactivity that they are deemed to represent an insignificant hazard to humans and the environment and are hence below the concern of the regulators. It follows that such wastes can be managed as non-radioactive materials. The ‘non-exempt’ categories of wastes need authorization for release based on limits set by the competent authority in consonance with national radiological protection criteria. Such criteria are based on recommendations issued periodically by the International Commission on Radiological Protection (ICRP). General guidelines and standards on radiation protection are worked out by other international organizations, including the IAEA, the World Health Organization (WHO), the International Labour Organization (ILO) and the Nuclear Energy Agency (NEA) of the OECD. National authorities promulgate appropriate radiation protection regulations based on these and other recommendations and monitor releases of radioactive materials of all types from nuclear facilities to ensure that they are observed [1].

Radiation and radioactive substances are natural and permanent features of the environment, and thus the risks associated with radiation exposure can only be restricted, not eliminated entirely. Additionally, the use of human made radiation is widespread. Sources of radiation are essential to modern health care: disposable medical supplies sterilized by intense radiation have been central to combating disease; radiology is a vital diagnostic tool; and radiotherapy is commonly part of the treatment of malignancies. The use of nuclear energy and applications of its by-products, i.e. radiation and radioactive substances, continue to increase around the world. Nuclear techniques are in growing use in industry, agriculture, medicine and many fields of research, benefiting hundreds of millions of people and giving employment to millions of people in the related occupations. Irradiation is used around the world to preserve foodstuffs and reduce wastage, and sterilization techniques have been used to eradicate disease carrying insects and pests. Industrial radiography is in routine use, for example to examine welds and detect cracks and help prevent the failure of engineered structures.[2] These activities inevitably produce radioactive waste, once the useful span of the sources come to an end.

2. Classification of wastes

Radioactive wastes can be classified in different ways: according to source, to its form (i.e. solid, liquid or gaseous), to radioactivity levels, to amounts of long lived and short lived radionuclides, to intensity of highly penetrating radiation, to final disposal requirements or to toxicity.

Classification based on half-life leads to two categories: “short lived” and “long lived”. This is consistent with the disposal route. Long lived radioisotopes may be disposed off through
burial/geological disposal, whilst less stringent disposal routes may be used for short lived isotopes (delay and decay).

Classification based upon concentration leads to definition of high level, intermediate level and low level wastes (HLW, ILW, LLW). The distinction made here is on the basis of shielding requirements during handling, toxicity of the radionuclides involved and possibly, also the amount of heat generated by the waste due to radioactive decay.

Low level wastes contain negligible amount of long lived radioisotopes (e.g. tritium). Intermediate level wastes describe wastes with significant beta/gamma activity and low alpha activity. High level wastes are highly radioactive, heat generating and long lived. Also alpha bearing wastes include wastes contaminated with significant amounts of long lived, alpha emitting nuclides.

All categories of wastes may or may not contain chemotoxic materials which may be stable elements (heavy metals) or degradable organic materials. Such wastes are called “radioactive mixed wastes” (RMW).

Wastes contaminated with only short lived radioisotopes, or containing very low concentrations below authorized levels for release into the environment, do not need treatment or conditioning. For example, iodine – 131 used orally for diagnosis and treatment of cancer has a half life of 8 days. Its retention for 64 days or 8 half lives reduces its activity to less than 1% of the original activity.

Long lived radioisotopes may need treatment and conditioning. The purpose of which is to make the material suitable for its subsequent management, which includes storage, transportation and final disposal. Treatment involves volume reduction. Conditioning adapts the treated waste into a form suitable for storage and eventual disposal in a selected environment. In Kenya the treatment and conditioning of solid wastes (radioisotopes) is well defined, while those for liquid and gaseous state are yet to be developed.

3. Sources of wastes and their management

Wastes are generated from various applications:

In research these include malaria control, sleeping sickness caused by the tse-tse fly, tripanosomiasis, nutrition, primate research, livestock, insects. These involve the use of small quantities of radionuclides used as tracers to follow the fate of certain chemicals or chemical elements. These produce low-level long-lived wastes as most activity are in the order of microcuries. These are in the form of solid waste (contaminated gloves, syringes, vials, waste paper, etc) and a mixture of liquids that are stored in plastic drums in various research institutes awaiting further treatment.

Industrial sources like moisture density gauges are used in the horticultural and in the road construction industries. The sources commonly used are americium-241/beryllium. These are long lived isotopes and up to now have not been a major problem regarding radioactive waste management. Several iridium-192 sources are used in industrial radiography for non-destructive testing of welds among others. Other gauges used are strontium-90 in the paper and tobacco industries and as level gauges in the beer industry. These are long lived radioisotopes and, after use, are dealt with as solid wastes.

In medicine, cobalt-60 sources used in radiotherapy departments in hospital apply a “return to manufacturer” policy and therefore waste in this area is not a problem. Cesium-137 in brachytherapy is dealt with as solid waste. For iridium-192 and iodine-131 the method of delay and decay is applied. The same as the technicium generator for nuclear medicine. BACTEC reagents used for shortening period used to diagnose tuberculosis also generate volumes of solid waste similar to those from research institutions.

Solid sources from schools and colleges in the order of microcuries are used as demonstration aids in science laboratories. These are eventually managed as solid wastes.
4. Conclusions

In Kenya, at the moment, solid wastes (radioisotopes) are immobilized in concrete matrix using 200 litre drums and stored in a temporary shed whilst awaiting final repository once appropriate land is found and acquired. At the moment eight drums are stored this way and their contents are well documented.

The country is still facing problems in technology for managing liquid waste. Most liquid wastes are generated by research stations, who store them safely in their backyards awaiting a way forward, usually in RMW. The solid wastes from such institutions and carbon–14 wastes from hospitals have in the past been incinerated, but lobby groups and the introduction of the National Environmental Management and Co-ordination Act (NEMA) recently has changed this. Such wastes are now also stored by the respective waste producers. Expertise in treatment and conditioning of such waste and the technology is still lacking and we hope to tackle this problem through a new country project with the IAEA.

Another challenge is that the Board shall soon have to monitor protection of the environment from airborne dust and the waste management of monazite from the mineral sands industry. The Board hopes to tackle this with assistance from the IAEA. It may involve familiarization of at least one regulator with a mining site with well developed technology and mining infrastructure through a scientific visit.

The Radiation Protection Board is the regulatory authority and oversee the safe use of ionizing radiation in the Country. The Board also enforces the Radiation Protection Act, CAP 243 Laws of Kenya. RWM Regulations have been developed and now await enforcement.

REFERENCES


Host rock tomography for radioactive waste disposal research in Hungary

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Hungary

Abstract. This paper represents a summary of the current research programmes on radioactive waste disposal in Hungary and shows results of tomographic survey at the Morágy Granite Formation as the potential site for low- and intermediate-level radioactive waste disposal. The paper was compiled from the reports of the research programmes of Mórágy Granite Formation and Boda Claystone Formation, managed by the Public Agency for Radioactive Waste Management (PURAM).

1. Introduction

According to Article 40 of the Act No. CXVI. on Nuclear Power from 1996, the Government is liable to execute the tasks of final disposal and interim storage of Hungarian radioactive wastes and decommissioning of the nuclear facility. The related activities are financed by the Central Nuclear Financial Fund, which is a separated state financial fund. The manager of the Fund is the Hungarian Atomic Energy Agency, which established, the Public Agency for Radioactive Waste Management (PURAM), to accomplish the related tasks. The ongoing research programmes include two issues: the near surface facility for low and intermediate level radioactive wastes (L/ILW) in the Morágy Granite Formation (MGF) and the deep geological disposal of spent fuel (SF) and high level radioactive wastes (HLW) in the Boda Claystone Formation (BCF), both carried out and financed in form of mid- and long-term plans and investigation programmes.

2. Waste inventory of HLW and conditioned L/ILW

The sources of the national radioactive waste are diverse. The highest amount of waste is produced by the only one Hungarian nuclear power plant, the Paks Nuclear Power Plant (Paks NPP), with its four VVER–440 reactors, generating approximately half of the nation’s electricity. Paks NPP was planned to produce energy for 30 years. Due to its important role in the Hungarian energy production and the lack of substitutive sources, a 20-year period of extension in the operation is planned but still not decided.

Small amounts are produced by research facilities like the training reactor of the Budapest University of Technical and Economical Sciences, Institute of Nuclear Technology and the research reactor in the Atomic Energy Research Institute of the Central Physical Research Institute.

Table 1. The amount and origin of Hungarian nuclear waste

<table>
<thead>
<tr>
<th>WASTE TYPE</th>
<th>30 yrs operation time</th>
<th>50 yrs operation time</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>HLW [1]</td>
<td></td>
</tr>
<tr>
<td>spent fuel (Paks NPP)</td>
<td>11 266 pieces</td>
<td>18 706 pieces</td>
</tr>
<tr>
<td>spent fuel (other sources)</td>
<td>3 225 pieces</td>
<td>3 225 pieces</td>
</tr>
<tr>
<td>operational wastes</td>
<td>173 m³</td>
<td>263 m³</td>
</tr>
<tr>
<td>decommissioning wastes</td>
<td>247 m³</td>
<td>247 m³</td>
</tr>
<tr>
<td>Püspökszilágy RWTDF a</td>
<td>100 m³</td>
<td>100 m³</td>
</tr>
<tr>
<td>conditioned L/ILW [2]</td>
<td></td>
<td></td>
</tr>
<tr>
<td>solid waste</td>
<td>2 547 m³</td>
<td>ND</td>
</tr>
<tr>
<td>ion exchange synthetic resin</td>
<td>639 m³</td>
<td>ND</td>
</tr>
<tr>
<td>evaporation residue</td>
<td>16 067 m³</td>
<td>ND</td>
</tr>
<tr>
<td>other liquid wastes</td>
<td>1 649 m³</td>
<td>ND</td>
</tr>
</tbody>
</table>

a Radioactive Waste Treatment and Disposal Facility
3. The research projects

(a) The BCF programme for the final disposal of spent fuel and high-level radioactive waste started as a preliminary characterization programme between 1989-1992, which was followed by the Alfa Project (1993-1995) and the Short Term Project (1996-1999). These latter two projects were carried out underground in the so-called Alfa shaft, which reached the BCF through an investigation shaft driven from the sandstone block of the former uranium mine near the city of Pécs. The technical co-ordinator of the underground research was MECSEKÉRC Ltd., the official controller was PURAM. The results were summarized in a 10-volume report published by MECSEKÉRC Ltd. in 1998. In 1999 the shafts and tunnels of the uranium mine were filled back, therefore the access to the Alfa shaft was blocked irrecoverably. The two phase (the site selection and the site characterization phase) Middle Term Project, started in 2003, is planned to have been completed by 2008.

(b) The MGF programme for the final disposal of low- and intermediate-level radioactive waste started with site selection in 1993 and ended in 1996. The results of this period were published in the Annual Report of the Geological Institute of Hungary (MÁFI), 1996/II and were followed by the site characterization running till 1999. The ground-based exploration was carried out in 2002-2003 under the responsibility of Bátatom Ltd., a consortium of four institutes and companies: Geological Institute of Hungary (MÁFI), ETV-Erőterv, Golder Associates Hungary and MECSEKÉRC Ltd.

4. The main goals of the MGF research programme

The research activities for the subsurface investigation phase started in 2004 have different goals:

(a) to find and locate a suitable rock body for the repository. This means that the investigations focus on the expected repository level after the establishment the inclined tunnels. To reach the goal all necessary measures shall be taken, including additional studies from the ground surface;

(b) to characterize the selected rock to provide data for the design and construction of the repository and the safety analysis. It is necessary to have a preliminary layout to position the characterization boreholes in a reasonable way to avoid disadvantageous hydraulic connections within the repository volume;

(c) to better understand the geology, tectonics, geotechnics and hydro-geology of the site. The access tunnels will provide a good opportunity to study these topics in the actual environment. Their construction can be regarded as a training phase to learn how the rock mass actually behaves and influences the surroundings.

5. Tomographic survey of MGF at Bátaapáti

The complementary geophysical survey at Bátaapáti was carried out in the scope of the geological exploration aimed at the final disposal of low- and intermediate-level radioactive waste. The measurements were related to the drilling activity, in 2002-2003, and were made in the central part and its closely connected area. The conditions of the area from a geophysical aspect are unfavourable. The physical parameters of the loess, which is a 40-60 m thick cover on the granite body, makes it hard to apply both electromagnetic and seismic methods to investigate the granite mass and the Bátaapáti site.

The results of geophysics (absorption and velocity fields) are unusable without geological or hydro-geological explanation. Fortunately, examinations conducted by other methods in the area produced a large amount of data, thus facilitating the geophysical interpretation. The geophysical methods used did not image the geological structures directly but the physical variations of the rock were explored. Parameters measured by geophysical methods usually represent average values of a space domain determined by the resolution.

5.1. Seismic velocity and absorption tomography

Seismic tomography is an image reconstruction technique. If measured data are line integrals of the observed physical quantity, the distribution of the physical quantities of the inner structure can be determined from measurements carried out along the boundary of the given domain. Such a kind of connection between wave propagation types and the reciprocal of the velocity, and between the
logarithm of the amplitudes and the absorption, is known from seismic studies. The distribution of velocity and absorption can be determined by seismic tomography when the propagation times and amplitudes between shot points and geophones are measured along ray paths crossing each other. To get a reliable profile of adequate resolution the observed area must be covered uniformly by a multitude of rays in conformity with direction and number.

At the Üveghuta Site seismic tomographic measurements were carried out between adjoining pairs of boreholes in the technically executable depth ranges. To calculate the velocity propagation parameters in the granite only the data from the boreholes were used. If the sources or receivers were to be placed into the low-velocity loess layers the tomographic data system would be charged with considerable errors. This is because the thickness of the loess can be determined at only one cell precision and this time delay is comparable with the total runtime in the granite. The starting model and the boundary conditions for the SIRT (Simultaneous Reconstruction Technique) computer algorithms were provided by PSQ and PQ seismic borehole data (where P and S are seismic body waves, Q is the quality factor). The computation is based on the modification of the wave propagation parameters along ray-paths, which cross each other in the space domain between boreholes until the misfit between computed and measured parameters is minimal.

The resolution of tomography between boreholes is direction-dependent, especially in the case of large borehole distances because of the partial absence of near vertical rays; consequently steep elements are not imaged. Another inherent characteristic of imaging is that the accurate velocity of a small-sized, low-velocity structure is not mapped adequately by the tomography: it is “smeared” because the rays do not cross the given structure (Fermat’s principle).

The results of the tomographic measurements at the site show some spots or stripe-like low- or high-velocity granite bodies. The structures are considered to be 2D because of the lack of 3D data. Most of the low velocity bodies can be observed at rather shallow depths. In boreholes Üh-23-Üh-2-Üh-22-Üh-3, where the geometry was the most favourable, the velocity and absorption tomographic sections resolved even steep dipping elements [3] [4]. It can be inferred from resolution parameters that tomographic spots and forms of zones do not necessarily display the peculiarities of the parameters recognized in drill-core or well logs. With this method changes comparable to the wavelength can be observed. These changes are caused by the granite material, the fissures in the granite, the fissure infillings, the direction of fissures etc.

Experiences at the Üveghuta Site show that seismic velocity and absorption are less affected by the rock stresses and the rock material, but they depend definitely on the rock-mechanical conditions.

The results of velocity and absorption tomography should be interpreted together: their data along the boreholes are in good correlation with smoothed, averaged well-log data principally with electric resistance, acoustic and seismic velocity sections.

5.2. Imaging the granite inside

Geophysical information with the best resolution is provided by the well-logging applications because they made measurements in the immediate vicinity of the observed material. In spite of the good resolution of the methods, the measured parameters do not generally correlate, even in closely spaced boreholes. For the macro-level spatial description of the extent of the granite body, the number of boreholes and the interpretation of well-logging data are insufficient altogether. There is a special character of the granite body beyond what can be observed in the boreholes and that is the variability in the scale between boreholes. This can be imaged with the best resolution by seismic cross-hole methods.

Lacking other possibilities the tomographic results were evaluated as phenomena with clear changes in the plane of the boreholes, while the changes of the actual granite show 3D features. This is verified by seismic tomographic sections which were measured in the nearby planes of Üh-3-Uh-23, and Üh-3-Üh-22-Üh-2-Üh-23. Comparing the suitability of methods for extension of the attributes to the area inside and between the boreholes, the magnetotelluric method, 3D seismic first break tomography and S-wave reflection profiling are suitable from point of view of the order of resolution at the investigated
area. The integrated interpretation of the borehole tomography outlined the inner structure of the granite. The granite bodies are defined by seismic tomography (Fig. 1).

**FIG. 1: Absorption tomographic section between boreholes**

6. **Conclusions**

The methods described contributed fundamentally to the investigation of the surface and to the knowledge of the spatial characteristics of the granite. The granite in the boreholes, and in its extended surroundings, can be well characterized by seismic tomography. The weakened zones between the bodies could have importance from the point of view of water conduction and their roles need further investigation.

**REFERENCES**


Current Status of radioactive waste disposal in Japan

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Abstract. The Nuclear Safety Commission of Japan (NSC) has issued a basic policy for safety regulations and the safety guidelines concerning disposal of solid radioactive waste and decommissioning of nuclear facilities. Based on these policies and guidelines the regulatory bodies, Ministry of Economy, Trade and Industry (METI) and Ministry of Education, Culture, Sports, Science and Technology (MEXT) have carried out the practical activities such as amendment of safety regulations. The NSC will set forward to develop the relevant safety guidelines of radioactive waste disposal and decommissioning of nuclear facilities.

1. Development of safety regulations for the disposal of high-level radioactive waste (HLW)

Japan has a strategy of reprocessing of spent nuclear fuels. The high-level liquid waste generated from the reprocessing of spent fuels is immobilized in a glass matrix and stored for 30 to 50 years for cooling, and then will be disposed of in a stable geological formation deeper than 300 meters below.

The Specified Radioactive Waste Final Disposal Act (the Act) was promulgated in June 2000. In this Act the relevant items such as the establishment of implementing organization and selecting process of final repository are provided. Based on the Act NUMO, the Nuclear Waste Management Organization of Japan, was established as an implementing organization in October 2000. The NSC has published the Basic Policy of Safety Regulations on the Disposal of High-level Radioactive Waste (First Stage Report) in November 2000 [1] and published the Environmental Requirements to be considered in the Selecting Stage of Preliminary Investigation Areas (PIAs) of High-level Radioactive Waste Disposal in September 2002 [2]. The NSC’s Advisory Board on High-level Waste Repository Safety was established in September 2001 and the technical issues relating to the safety of HLW disposal have been investigated and discussed. In order to have the public opinion widely, the workshop concerning to the concept of the HLW disposal has been held four times to date.

In the Act, the final repository is to be selected in a step-wise manner by the three stages; that is, the initial “Preliminary Investigation Areas” based on the literature survey and investigation carried out from the ground surface, then a reduced number of candidate site of “Detailed Investigation Areas” (DIAs) and finally the repository site. NUMO commenced the open solicitation by sending a call for volunteers to all 3,239 municipalities in Japan in December 2002.

The NSC issued “A Commonly Important Issue for the Safety Regulations of Radioactive Waste Disposal” in June 2004. [3] In this report, the basic concepts such as radiation protection standard for the disposal of radioactive wastes, safety assessment by the risk-informed approach and compliance period were discussed. From now on, the environmental requirements for the selection of DIAs will be established. The design requirements for the repository, individual dose limit or safety indicators in the safety assessment, and so on will be discussed and a basic guideline for the safety examination will be prepared.

2. Development of safety regulations for the disposal of low-level waste (LLW)

In Japan, LLW are categorized into four groups according to their characteristics, and basic policy and safety guideline of each group are examined. The categorization of low level waste are as follows:

(1) radioactive waste generated from Nuclear Power Plants (NPPs).
(2) TRU waste generated from reprocessing facilities and MOX fuel fabrication facilities,
(3) Uranium waste generated from uranium enrichment facilities and fuel fabrication facilities,
radioactive waste generated from radioisotope utilization facilities, research institutes and research reactors.

For every group of wastes, three disposal options are established depending on their radioactivity level. The disposal options are as follows:

(a) near surface disposal without engineered barrier,
(b) near surface disposal with engineered barrier,
(c) sub-surface disposal with engineered barrier.

To carry out the near surface disposal (disposal option (a) and (b)) safely, the NSC issued the safety guideline[4] for the disposal of radioactive waste generated from NPPs. The regulatory body, METI established the regulations for these disposal options of the waste in accordance with NSC’s safety guideline.

The near surface disposal facility with engineered barrier for radioactive waste generated from NPPs is under operation at Rokkasho site. The concrete waste generated from the decommissioning of the Japan Power Demonstration Reactor (JPDR) was disposed of in Tokai site of Japan Atomic Energy Research Institute by the method of the near surface disposal facility without engineered barrier. It was already closed and moved to the institutional control period.

Relatively high radioactive wastes such as inner structures of nuclear power reactors are to be disposed of in a sub-surface disposal facility with engineered barrier at the depth from 50 m to 100 m below the surface. By using the examination drifts, the geological environment and groundwater around the candidate disposal site are investigated by the implementing company. The NSC is going to establish the safety guideline of the sub-surface disposal with engineered barrier.

For the disposal of wastes from radioisotope utilization facilities, NSC issued the basic policy of safety regulation for the near surface disposal of the waste except for disused sealed radiation sources. Based on this policy, the regulatory body, MEXT, has established fundamental criteria for the disposal. Safety regulations for disposal of TRU waste and uranium waste are remained not to be established.

### 3. Evaluation of clearance level

The commercial nuclear power plant, Tokai Power Plant of the Japan Atomic Power Co. has terminated of its operation and its first stage of decommissioning has already been initiated in March 1998. Decommissioning of other commercial power plants will be followed successively in the near future.

In order to recycle and reuse of large amount of slightly contaminated concrete and metal waste generated from the decommissioning of NPP, NSC has discussed and examined to establish the clearance level regulations of such radioactive wastes since 1997.

Constructing the assessment scenarios taking into account of the Japanese lifestyle, the clearance levels of 58 radionuclides included in concrete and metal wastes generated from the decommissioning of nuclear reactors and of the post-irradiation examination (PIE) facilities were evaluated.

IAEA proposed exemption levels for bulk amounts of material in the Safety Guide No.RS-G-1.7 in 2004 [5]. It is described that the proposed levels presented in this Guide may be used by the regulatory bodies of each Member State as a basis for the clearance. NSC reviewed the clearance levels by adopting several proposed evaluation concepts in RS-G-1.7; for example, exposure for children (1-2 ages), skin exposure [6]. As the result of review, no significant difference between NSC clearance levels and IAEA exemption levels was recognized. So the NSC suggests from the viewpoint of consistency with international standards that the IAEA’s exemption levels may be appropriate to choose as the Japanese clearance levels for concrete and metal waste generated from the decommissioning of nuclear reactors and PIE facilities. Regulatory bodies have followed the procedure of formulating the clearance level in the existing regulations at present.

Clearance levels are calculated as activity concentration assuming that the solid material having a mass of several tons is uniformly contaminated. However, actual contaminated material has not uniformly distributed in the waste form. In establishing the verification methods, it is necessary to take into
account the radioactivity distribution in the waste form. The regulatory bodies will approve the verification process for clearance level and the NSC intends to confirm the relevance of the process.

4. Safety regulation system of decommissioning

Decommissioning of NPPs and research reactors involves the removal of radioactive material, including spent fuel. After the shutdown of such reactors, these removal activities reduce the radiological hazards and the inventory in the installation. So the regulatory system should be carried out on a case by case basis under the progressive and systematic reduction of regulatory control in radiological hazard, paying attention to the decommissioning activities. The NSC has investigated the existing regulatory system for decommissioning of nuclear power plants and the relevant facilities and made some suggestions to the METI to the existing regulation [7]. The METI is now trying to review and revise the regulation.

The standard for site release from the regulatory control will be developed henceforth. In establishing the site release standard, the dose rate of public exposed by the reuse of land after the release of the site, approach of safety assessment and methods of measurement of residual radioactive contamination will be the important issues. These issues will be considered and developed referring to and based on the relevant activities of IAEA and international trend.

5. Future work

The NSC set forward to develop the safety guideline of sub-surface disposal with engineered barrier for reactor-core components and the basic guideline for the safety examination of HLW referring to national and international R&D results.

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Management of decommissioning low level radioactive wastes and clearance materials in Japan

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Abstract. For a commercial nuclear power plant, MITI (Ministry of International Trade and Industry, currently METI) recommended (June 1985) the following standard decommissioning process: system decontamination, safe storage for 5~10 years, dismantling and removal. The decommissioning project on the first commercial power reactor, the Tokai power station of JAPCO, the Japan Atomic Power Co. was started in December 2001. The project will take 17 years and is therefore divided into three phases, and reactor structure will be dismantled at the third phase from 2011, after 10 years safe storage. At present, conventional facilities are removed, and radioactive wastes raised from these activities are of small amount and they are stored at the Tokai site. All radioactive wastes raised from decommissioning of the commercial nuclear power plants are classified as low level radioactive wastes in Japan. Those are divided into 3 categories from the viewpoint of disposal, in other words, 3 different types of disposal facility will be constructed. The Atomic Energy Commission (AEC) and the Nuclear Safety Commission (NSC) have concluded a disposal policy and basic regulatory concept of disposal of these wastes. Relatively high radioactive concentration waste, such as reactor internals, with relatively higher contamination, will be disposed of at 50-100m depth, so called intermediate depth repository. Based on the concept, utilities and Japan Nuclear Fuel Ltd. (JNFL) are carrying out basic study and site investigation. With respect to very low level radioactive wastes, whose radioactive nuclide concentration are exceeding clearance level, JAPCO started investigation on site characterization, to gather data on ground water level and flow rate, which will be utilized for safety assessment of VLLW disposal facility. NSC reviewed clearance level based on RG-S-1.7 “Application of the concept of Exclusion, Exemption and Clearance” issued by the IAEA. Establishment of regulations of clearance is now forwarded by NISA. JAPCO is preparing clearance measuring devices and rules of quality assurance. Public acceptance is also a very important issue for non-restricted use. The Tokai-1 decommissioning project plays an important role in demonstrating that the decommissioning of commercial nuclear power plant can be executed safely and economically, and for establishing the key technologies for future LWR decommissioning in Japan.

1. Introduction
Tokai-1 nuclear power plant (Gas Cooled Reactor) of JAPCO started commercial operation in 1966 as the first commercial nuclear power plant in Japan and ceased its operation in 1998. Spent fuel elements were removed out of the reactor core and shipped to the reprocessing plant shortly after the termination of operation, and these de-fuelling activities were completed in June 2001. JAPCO launched Tokai-1 decommissioning in December 2001 after the submission of the notification of decommissioning plan to the competent authorities. This is the first instance of the decommissioning of a commercial nuclear power plant in Japan. As the whole project is planned to take a long term (17 years in all), the project program is divided into three phases. Now JAPCO is carrying out the first phase of decommissioning work, and also preparation for the monitoring for compliance with clearance level.

2. Project Schedule and Present Status
According to METI’s standard decommissioning process, JAPCO’s strategy on Tokai-1 decommissioning project is that the Tokai-1 plant would be dismantled continuously through three phases (stages) and the land will be a green field for future nuclear power generation as shown in Table 1. The reactor area, i.e. reactor and biological shield envelope, will be stored in safe condition for 10 years to reduce radioactivity.
Prior to the reactor dismantling, conventional facilities outside the reactor area are to be removed for the purpose of securing a transportation route for reactor dismantling wastes, and also to get the space for new waste treatment facilities. These conventional facilities dismantling work would balance the workload through the 17 years long term decommissioning project.

Table 1: Decommissioning project schedule

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>First Phase</td>
<td>Preparation work</td>
<td>Remove conventional Facilities</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Second Phase</td>
<td></td>
<td></td>
<td>Remove SRUs</td>
<td></td>
</tr>
<tr>
<td>Third Phase</td>
<td>Safe-Store of Reactor Area</td>
<td>Reactor Dismantling</td>
<td></td>
<td>Building Demolition</td>
</tr>
</tbody>
</table>

The First Phase takes 5 years from 2001 to 2005. The first activity was preparation for the reactor safe-store; i.e. all primary loop’s valves connecting to the reactor were closed in 2001. Then the decommissioning preparatory works such as the draining of and cleaning of the cartridge cooling ponds (Fig.1), the modifications of the electrical power supply facilities, etc have already been completed. Equipment of the Turbine building and the Reactor service building have already been removed. The removal activities of equipment in the Fuel handling building and Fuel charge machine are now under way (Fig2).

3. Waste treatment and disposal

All radioactive wastes from the Tokai-1 decommissioning besides spent fuel reprocessing wastes are classified as the Low Level radioactive Waste (LLW) in Japan, and the LLW is further categorized into three classes in view of burial disposal. The amount of wastes arising from Tokai-1 decommissioning is estimated at 192kt in total, and about 10% of them are estimated as radioactive wastes, as shown in Table1. Radioactive wastes are treated (decontamination, melting, compaction, and so on) and packaged in containers. Eventually they will be disposed of at burial facilities in accordance with their radioactive level. The amount of wastes arising in the first and the second phase is small and the wastes are stored in the existing storage facilities on Tokai site until the commencement of third phase. The burial disposal facility is expected to be determined before the commencement of the third phase (reactor dismantling); the majority of wastes arises in this phase.
Table 2: Estimation of the waste amount from Tokai-1 decommissioning; unit: [kiloton]

<table>
<thead>
<tr>
<th>Classification</th>
<th>Main Subject</th>
<th>1st Phase</th>
<th>2nd Phase</th>
<th>3rd Phase</th>
<th>Total</th>
<th>Disposal</th>
</tr>
</thead>
<tbody>
<tr>
<td>Low Level Waste</td>
<td>Comparatively high radioactive level [L I]</td>
<td>Graphite blocks, Core component</td>
<td>0</td>
<td>0</td>
<td>1.55</td>
<td>1.6</td>
</tr>
<tr>
<td></td>
<td>Comparatively low radioactive level [L II]</td>
<td>Reactor pressure vessel, Shield concrete</td>
<td>0.01</td>
<td>0.56</td>
<td>7.84</td>
<td>8.5</td>
</tr>
<tr>
<td></td>
<td>Very low radioactive level [L III]</td>
<td>Steam raising units, Shield concrete</td>
<td>0.01</td>
<td>0.06</td>
<td>8.01</td>
<td>8.1</td>
</tr>
<tr>
<td></td>
<td>(Sub total)</td>
<td></td>
<td></td>
<td></td>
<td>18.1</td>
<td></td>
</tr>
</tbody>
</table>

Non-Radioactive waste (Including Clearance waste)

Total: 11 8 173 192 -

The amount is after decontamination.

Graphite blocks and some of the core internals are L1 waste, and those wastes shall be disposed of at an intermediate depth (50~100m) disposal facility. Japan Nuclear Fuel Limited (JNFL) is now carrying out detailed site survey and basic design.

Reactor pressure vessel, bio-shield concrete and spent fuel pond concrete are L2 waste, and they shall be disposed of at a near surface (with artificial barrier) disposal facility. We are now investigating on disposal facility for the waste raised from decommissioning, from viewpoints of difference of shape and configuration of waste package and composition of radioactive nuclide.

The other concrete and steel from steam raising units are L3 waste, and they shall disposed of at a near surface (without artificial barrier) disposal facility. We intend to construct L3 disposal facility on our Tokai site, and we started preliminary site survey for observation of ground water level and geological characteristics in 2004. The result of this preliminary site survey shows possibility to construct L3 disposal facility in our Tokai site, therefore we started detailed site survey in 2005.

3. Clearance level

Clearance materials are arising from early 2005 by Tokai decommissioning. Since RG-S-1.7 ‘Application of the concept of Exclusion, Exemption and Clearance’ issued by IAEA, NSC has reviewed clearance level and ‘Law for regulation of nuclear source material, nuclear fuel material and reactors’ has been amended in 2005, and related regulatory flame will be established by METI. At the same time, JAPCO is carrying out preparation work for clearance, such as investigation on measuring device, material handling procedure, quality assurance procedure and so on, to make it possible to bring out the clearance materials from restricted area soon after the establishment of related regulations. To achieve unrestricted recycling, public acceptance is important, therefore step-by-step approach is planned, i.e. usage of clearance materials by the electric power industry is to be done as the first step, to demonstrate the safety. For instance, JAPCO intends to fabricate steel bars for reinforcement of concrete, and to utilize them at construction of nuclear related concrete structures and so on.

4. Conclusion

Decommissioning of the Tokai-1 nuclear power plant, the first decommissioning of a commercial nuclear power plant in Japan, is currently under way on schedule. In addition to that, the project plays a role in accelerating the process towards the establishment of regulatory frame-work of decommissioning, LLW disposal, clearance and also accelerating the process towards the construction of LLW disposal facility. Keeping good relationship with local communities is also an important issue, and JAPCO always discloses information as much as possible to local government and residents.
Overview of mining and milling tailing and its environmental consequences in Nigeria

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Abstract. Materials of natural origin such as ore and minerals often contain high level of radioactivity caused by radionuclides of uranium and thorium series. Generally, mining operations are usually accompanied by dumping of large quantities of tailing in the environment. Tailing are rich in radioactive minerals and are dumped at close proximity. Prominent mining and milling tailing in Nigeria is as a result of several years of mining and processing of casseterite (tin ore) and (niobium ore) columbite, mining, crushing and processing of coal and phosphate rock. Of environmental importance are the tailing from mining, milling and processing of casseterite, columbite, coal and phosphate. Extraction and processing of these mineral produce tailing which are contain significant amount of radionuclide. This paper gives a review of tailing in Nigeria and its environmental impact.

1. Introduction

Mining and related activities contribute not in little amount to the workforce of the Nigeria. The economic effects of these activities are considered of high importance with little or no regard to the consequent environmental implications. Natives and small-scale enterprises usually dominate mining activities in Nigeria. The informal operators use primitive mining and processing techniques of digging and panning. Magnetic and electrostatic separator are usually employed by the small scale mill occupying area approximately 100 x 100m with a mill shade of 20 x 20 m size, an office block and the rest of area occupied by tailing and heaps accumulated over the years [1,2].

The trailing is considered non-hazardous because of the technique of mining and milling, the mining and milling waste are regarded as natural occurring radioactive materials (NORMs). The volume of tailing from local mining activities produced annually could reach a level so high that existing low level radioactive waste (LLRW) facilities would be readily occupied by NORM, if control disposal procedure is not adopted. These radioactive materials cannot just be ignored as being below radiological concern (BRC) or lower than the exemption concern level (ECLs). The IAEA [3] and the French Commissariat a l’Enerie Atomique [4] have already established exemption concentration level (ECLs) which are quite similar for natural occurring radionuclide. ECL are derived from annual dose, which is a small fraction of annual average received from a natural source of radiation by use of time dependent environmental pathway scenarios and dosimetric models.

Radiation exposure resulting from these abandoned tailing are not different from those observed from over the world. This results in contamination of vicinity [5,6] with radionuclide, contamination of the soil, ground water, biota and release of dust laden with radionuclides. Access to dump sites are not restricted, natives and informal miners have direct access and regular visit to these sites. Tailing dump sites of abandoned mines are generally not perceived as hazardous by member of the public, the tailing debris are used in intensively in construction and farming [5]. Schuler [7] report that raw materials and processed building products can vary greatly in radionuclide contents, reflecting their origin and geological conditions at site of their production. Animals also wonder on this sites grazing on pasture without restriction. WHO [8] and Health Canada [9] classified the consequences of exposure by human into carcinogenic and non-carcinogenic effects, based on radioecological risk by radiations of radionuclides isotopes and the chemical risk of heavy metals. Hazards resulting from radioactivity exposure include cancer (thyroid, breast, liver, leukaemia e.t.c.), chromosome aberrations, stillbirths, Down’s syndrome, respiratory diseases and liver tumours. Documented effects in other parts of the world with similar situation poses the same problems as in Nigeria. Major activities that produce tailing in Nigeria include the under listed:
2. Cassiterite and columbite (tin and niobium)

Tin mining in Jos plateau started in 1904, in mid 1920's more cassiterite discovery were made resulting in greater demand for more mechanized extraction techniques. These result in generation of grate amount of tailing, resulting in radioactive contamination of soil in this vicinity [5,6].

Determination of the radiological assessment of tin and columbite mining activities in Jos plateau and environ was conducted by Ibeanu [10] as indicated in Table 1, who [10] estimated the annual mean external effective dose around 100mSv y\(^{-1}\) for the tailing sites and about 10mSv y\(^{-1}\) for staying on contaminated site. These measured values are in accordance with the external dose rate performed in twelve different Malaysia [11] and Thai plant [12].

Table 1: Mean measured dose rates and annual mean external exposure

<table>
<thead>
<tr>
<th>Sample</th>
<th>Measured dose rate ((\mu \text{Gy m}^{-1}))</th>
<th>Annual effective dose (mSv y(^{-1}))</th>
</tr>
</thead>
<tbody>
<tr>
<td>Contam. soils</td>
<td>7.4 (5.6 - 10.8)</td>
<td>9.1 ± 1.4</td>
</tr>
<tr>
<td>Tailing sites</td>
<td>81.5 (58.9 - 102.7)</td>
<td>99.9 ± 4.8</td>
</tr>
<tr>
<td>Control</td>
<td>0.1 (0.05 - 0.21)</td>
<td>0.12 ± 0.01</td>
</tr>
</tbody>
</table>

The mean values from Table 1 [10] exceed the dose limit recommended by ICRP 60 [13] and IAEA [15]. Since access to these sites are not restricted, the local populace spend time and use this radiation materials. Annual exposure for worker in the processing mill ranges between 2-180mSv y\(^{-1}\) [1] far above the recommended 20mSv y\(^{-1}\) [13,15].

3. Phosphate rock (phosphate mining)

Phosphate rock constitutes the bulk raw materials for the manufacture of phosphate fertilizers and some phosphate base chemicals. Study by Ogunleye and his group [14] revealed that phosphate rock is prominent in the Southeast section of Illellemmeden basin, Sokoto state, and in parts of Ogun state, North-West and South-West Nigeria respectively have been documented by the Geological Survey of Nigeria and several workers [16, 17]. Analysis by Ogunleye et al [14]. Table 2 indicated the rock contain enriched significant amount of uranium and thorium with average concentration of some toxic elements like As, Sb, Cr and Zn not in appreciable difference from agricultural soils. Guimond and Hardin [18] also report that phosphate rock generally have high concentrations of \(^{226}\)Ra daughter of uranium decay.

Table 2: Elemental concentrations of Sokoto and Togolese phosphate rocks.

<table>
<thead>
<tr>
<th>Element</th>
<th>Sokoto (Mean ± SD)</th>
<th>Togo (Mean ± SD)</th>
<th>Element</th>
<th>Sokoto (Mean ± SD)</th>
<th>Togo (Mean ± SD)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Al %</td>
<td>1.45±0.03</td>
<td>0.74±0.15</td>
<td>Mo</td>
<td>76±2</td>
<td>41±2</td>
</tr>
<tr>
<td>Ca %</td>
<td>46.9±0.4</td>
<td>52.7±0.6</td>
<td>Ba</td>
<td>397±45</td>
<td>302±29</td>
</tr>
<tr>
<td>Fe %</td>
<td>2.1±0.03</td>
<td>0.3±0.01</td>
<td>Cl</td>
<td>—</td>
<td>1298±90</td>
</tr>
<tr>
<td>Na</td>
<td>2300±10</td>
<td>1086±10</td>
<td>I</td>
<td>—</td>
<td>0.22±0.02</td>
</tr>
<tr>
<td>Mg</td>
<td>—</td>
<td>224±4</td>
<td>Cs</td>
<td>0.40±0.3</td>
<td>—</td>
</tr>
<tr>
<td>Mn</td>
<td>5716±45</td>
<td>149±2</td>
<td>As</td>
<td>11</td>
<td>9.07</td>
</tr>
<tr>
<td>Sc</td>
<td>11.8±0.3</td>
<td>11.6±0.1</td>
<td>Sb</td>
<td>0.4±0.03</td>
<td>0.6±0.1</td>
</tr>
<tr>
<td>Tb</td>
<td>0.7±0.1</td>
<td>—</td>
<td>La</td>
<td>243±0.4</td>
<td>120±1</td>
</tr>
<tr>
<td>Co</td>
<td>19.8±0.4</td>
<td>1.2±0.05</td>
<td>Ce</td>
<td>474±2</td>
<td>225±2</td>
</tr>
<tr>
<td>Cr</td>
<td>28±2</td>
<td>75±2</td>
<td>Nd</td>
<td>233±17</td>
<td>100±11</td>
</tr>
<tr>
<td>Hf</td>
<td>1.5±0.1</td>
<td>0.5±0.06</td>
<td>Sm</td>
<td>56±1</td>
<td>25±0.5</td>
</tr>
<tr>
<td>Zr</td>
<td>810±105</td>
<td>765±66</td>
<td>Eu</td>
<td>13.8±0.2</td>
<td>7.6±0.1</td>
</tr>
<tr>
<td>Zn</td>
<td>59±5</td>
<td>143±97</td>
<td>Gd</td>
<td>146±9</td>
<td>56±6</td>
</tr>
<tr>
<td>V</td>
<td>65±3</td>
<td>68±1</td>
<td>Tb</td>
<td>9.4±0.1</td>
<td>3.8±0.2</td>
</tr>
<tr>
<td>Ti</td>
<td>—</td>
<td>259±41</td>
<td>Tm</td>
<td>22±1</td>
<td>14.2±1</td>
</tr>
<tr>
<td>Th</td>
<td>3.2±0.2</td>
<td>17.4±0.2</td>
<td>Yb</td>
<td>29±4</td>
<td>7.9±0.1</td>
</tr>
<tr>
<td>U</td>
<td>65±3</td>
<td>72±0.3</td>
<td>Lu</td>
<td>3.4±0.1</td>
<td>1.2±0.1</td>
</tr>
<tr>
<td>P %</td>
<td>15.1±0.01</td>
<td>15.4±0.01</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
4. Bituminous coal

Nigeria coal reserve was estimated at about 2.75 billion tones and it is widely distributed over 13 states of the federation. It is largely a bituminous type [19]. Presently mining is carried out at Enugu, Okoba, and Omaraku. Like all ore, coal is also associated with natural occurring radioactive elements due to the content of $^{238}\text{U}$, $^{232}\text{Th}$ and $^{40}\text{K}$ [18]. This certainly has radiological implications not only for the miners but also for the populace in the immediate environment of the mines and the users. Balogun et al., [19] determined the activity concentration of $^{238}\text{U}$, $^{232}\text{Th}$ and $^{40}\text{K}$ from samples collected from Enugu. The activity concentration of radionuclides determined ranged from $0.02\pm0.002$ to $48.42\pm5.32$ BqKg$^{-1}$. Overall natural radionuclide contribution to the radioactivity of the environment was found to be $404.16\pm23.44$ BqKg$^{-1}$. Major contribution by coal tailing 49.5% of the above value. The outdoor and indoor exposure rates in air 1 m above the ground are estimated to be $(6.31\pm1.20)\times10^{-8}$ and $(7.57\pm1.20)\times10^{-8}$ Gy h$^{-1}$ respectively, for the mining environment. The resulting annual effective dose equivalent estimated is $(4.49\pm0.74)\times10^{-4}$ Sv yr$^{-1}$, this is below dose limit levels for workers recommended by ICRP 60 [13] and IAEA [15]. Comprehensive radioecology effects of coal mining in Nigeria still need to be done.

5. Conclusion

Death due to unknown causes and illness are usually experienced among local indigenes of Nigeria. One major cause of this may be as a result of radiation exposure. This cannot be adequately diagnose because community clinics available to the natives are not adequately equipped diagnose and treat radiation related illness. These makes it impossible to have an actual documentation of radiation related sickness and death. A comprehensive radiological assessment of the critical group, mining and related activities and need to be carried out to know the present level of radionuclide contamination and to be used for future monitoring references. Protection and relevant government agencies have much work to do in respect of inspection, regulation and enlightenment of people on the health implications of radiation exposure.

The licensing procedure for mineral exploration should adequately include estimate on volume of NORMS waste to be generated and disposal option to be adopted. This should include participation from all relevant government agencies, stakeholders and operators. Finally, remediation strategies need to be instituted to restore/reclaim land and to avert the continued and potential exposure to the local population including critical population group such as the children.

REFERENCES


Safe disposal and transportation of mining process waste (uranium ore) in Romania

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Romania

Abstract. The Romanian Nuclear National Plan (PNN) provides that radioactive materials (RAM), e.g. natural radioactive waste, is a subject of investigation for a safe and dedicated disposal as well as for a safe transport of the uranium ore and its concentrations taking into consideration its potential risks to human beings and the environment and possible radiological consequences due to this activity. The paper presents the main sources of natural radioactive wastes in Romania, the transportation routes of natural uranium ore from different subsidiaries to the CNU-Subsidiary Feldioara for processing. The on-site storage of the sterile at the uranium mine (at CNU- Crucea, Suceava Subsidiary) as well as the disposal in the conventional landfill sites (at CNU - Feldioara Subsidiary) of the very low-level waste resulting from handling, transformation and processing of the naturally occurring radioactive material, uranium ore, is presented. Specific problems related to the identification and evaluation of potential environmental risks and impacts as well as possible radiological consequences associated with the disposal of sterile and the very low-level radioactive waste resulting from processing of the uranium ore at Feldioara, Brasov are also approached. Also presented are possible risks associated with the uranium ore and uranium concentrates transportation in Romania both for routine and accident transport situations.

1. Sources of the mining uranium ore and its transportation routes in Romania

The Uranium National Company (CNU) from Romania is responsible, through other tasks, for: exploitation of the uranium ore, transport and processing of the uranium ore, concentrates processing and delivering of the final products, e.g. U\textsubscript{3}O\textsubscript{8} and UO\textsubscript{2}, and safety disposal and transportation of the processing waste. The company has different subsidiaries in Romania, among them are: Magurele, Bihor, Suceava and Feldioara. The routes for the transport of uranium ore are shown in Figure 1:

\begin{figure}[h]
  \centering
  \includegraphics[width=\textwidth]{routes.png}
  \caption{The routes for the uranium ore and concentrates in Romania}
  \label{fig:routes}
\end{figure}

The uranium ore is transported from Bihor and Suceava subsidiaries by railway to the CNU Feldioara for processing. Every special wagon has approximately 50 t of uranium ore. From uranium ore mine Crucea to the Vatra Dornei railway station (Suceava subsidiary), approximately 42 km, the ore is...
transported by road by means of special licensed trucks. Every truck is loaded with 13 tones of uranium ore. During the transport the contents of the uranium ore from truck is covered with a special tarpaulin. From uranium ore mine Baita to the Stei railway station (Bihor subsidiary), approximately 26 km, the ore is transported by road by means of special licensed trucks also. The uranium dioxide concentrate processed at CNU Feldioara (Brasov) is transported by road to the Nuclear Fuel Work, Pitesti.

2. Disposal of the uranium mill tailings at CNU Feldioara Brasov

Upon arrival at the receiving ore facilities at CNU Feldioara, the uranium ore form wagons is sorted out at a radiometric station. The activity of uranium ore will not exceed 32,222 GBq and the external dose at 1m from the wagon will be less than 0.02mSv/h. After processing within Feldioara nuclear facilities, the resulting uranium mill tailings are transported by means of a Ø 219 mm pipe, approx. 2 km, to a mud-setting pond-b Cetatuia 2, compartment 2. In this sludge bed pond gravity settling decantation takes place. The solid uranium mill tailings fall at the bottom of the pond and the liquid part is transferred through a pumping system in the Mittelzop sludge bed. The uranium waste has an uranium content of about 6mg/l and an activity of about 152.1 Bq. Again, from this liquid part, the uranium is recovered through a special procedure, and the remaining liquid is transferred to the Olt river. The maximum uranium content of the waste transferred into the waters of the Olt river will not exceed 0.6 mg/l. The restriction (pass criteria) for uranium content of the waters of the Olt river is less than 0.021 mg U/l.

3. Disposal of the sterile uranium ore at Crucea uranium mine

The uranium ore extracted from the uranium mine Crucea is radiometric sorted and if the quantity of the ore has less than 0.004% uranium content this is considered sterile and is deposited on special conventional landfill sites. These places are strictly delimited as “controlled zone” which is checked regularly by the Nuclear Regulatory Body from Romania, CNCAN. The ore, with more than 0.02% uranium content is transferred in 3 special tanks. From these tanks the uranium ore is transferred into the trucks and transported to the Vatra Dornei railway station. Figures 2-6 show these activities:

FIG. 2: The special tank for uranium ore, Crucea uranium mine

FIG. 3: The special landfill site-sterile disposal, Crucea uranium mine
4. Identification and evaluation of potential risks due to the transport and disposal of the very low level radioactive waste

4.1. Transport by road

As shown in Figure 1 the routes for transport of the uranium ore and uranium concentrate are both by road and by rail modes. In order to evaluate the dose resulting from possible road accidents involving these radioactive shipments, based on the frequency of occurrence of accidents of specified severity, the computer code INTERTRAN II has been used. On the other hand for rail transport a probabilistic risk assessment method (PRA) has been adopted for this work aimed at quantifying the potential radiological consequences and the expected probability of occurrence of such accident sequences.
This assessment considers the transport operation and specific details (such as route and vehicle) which would be confirmed prior to a decision to undertake the movements.

Data to be used as input data to the computer code INTERTRAN II has been provided by postulate possible accidents such as: transport hazards (fixed impact hazard, mobile impact hazard), accident frequencies by road. Based on these there were calculated road accident probabilities. A summary of the accident probabilities for the road routes is presented in Figure 1:

- probability of impact only: $0.435 \times 10^{-5}$ per journey;
- probability of impact and fire: $1.53 \times 10^{-10}$ per journey;

It is also assumed that, following an impact, the content may become available for dispersion. The collective dose assessed areas follows:

- dose to public along route: $0.23 \times 10^{-5}$ person.Sv.y$^{-1}$;
- dose to public during stops: $0.32 \times 10^{-8}$ person.Sv.y$^{-1}$;
- dose to truck crew: $0.47 \times 10^{-5}$ person.Sv.y$^{-1}$

The total annual collective dose is: $0.80032 \times 10^{-5}$ person.Sv.y$^{-1}$. The associated latent cancer fatality risk is estimated at $0.76 \times 10^{-10}$ y$^{-1}$.

4.2. Transport by rail

There are different kinds of operation contributing to the overall risk, such as: rail transport, rail road transfer activities (from Crucea to Vatra Dornei and from Baita to Stei), handling and misoperation activities, etc. Transport and handling of possible accidents may occur and pose a risk to the public and the environment. Because the occurrence of such accidents is statistical in nature, probability risk assessment (PRA) has been adopted in order to quantify the potential radiological consequences and the expected probability of occurrence of such accidental sequences. The potential radiological consequences have been calculated by using the INTERTRAN II computer code. The calculated radiological risks include:

- radwaste exposure of the public and transport personnel from routine (incident free) transport of the very low level radioactive material (uranium ore);
- transport accident resulting in radiation exposure of the population and contamination of the environment.

The accidental sequences includes steps such as:

- characterization and the type and quantity of shipment;
- determination, selection and description of the type, severity and probability of occurrence of transport and handling accidents;
- assessment of potential radiological consequences for the spectrum of wealth condition encountered along the rail route.

The computer code INTERTRAN II has been used to determine the collective dose to population and transport personnel and the preliminary risk assessment results are:

- crew: $1.34 \times 10^{-5}$ person.Sv.y$^{-1}$;
- members of the public: $1.78 \times 10^{-5}$.

Total: $3.12 \times 10^{-5}$

Radioactivity releases are not expected to occur in close proximity to an possible accident site at a probability level as low as $10^{-7}$, i.e. a chance of 1 in 10 million for the total volume of the uranium ore to be transported.

4.3. Disposal of uranium mill tailings

The disposal option in mud-setting pond sites was chosen for very low-level waste resulting from handling or the transformation of naturally occurring radioactive material (uranium ore).

Whatever the waste level of activity to be considered, the intrinsic safety of the selected option is assessed through an approach without exemption or clearance levels, which is only based on
radiological impact studies demonstrating the special care and respect for human beings of the limit of exposure of the public of 1 mSv.person. y⁻¹.

The mill tailings present a low to very low activity which decreases very slowly. They owe their radiological characteristics to the natural radioactive elements initially present in the ore (mainly radium 226). Approximately the total sum of the daughters of uranium are present in mill tailings and the specific activity accounted is about 10⁵Bq.kg⁻¹.

The mill tailings are disposed at Feldioara CNU Subsidiary, near the place of production, in natural basins (mud-setting pond), valley closed by dam. All the disposals are managed by CNU Bucharest.

This disposal site was subjected to remediation. To prevent risks, and to waterproof and minimize at maximum the penetration of the radioactive waste liquids inside the underground waters, the bottom of the Cetatuia 2 mud setting pond has successive layers of: clay, high density polyethylene and bituminized cartoon. Also, the Mittelzop sludge bed pond is located on a former clay cavity.

The effectiveness of the containment device is checked by monitoring of the level of background radiation, radon concentration, quality of rejected water and, when necessary, measurements in the food chain are performed. Because the disposal site is closed by dams, monitoring of mechanical stability is performed.

CNU has written detailed procedures related to the remediation of the disposal site, among them are:

- disposal in conventional landfill sites is the best solution of the efficient management of the mill tailings;
- radiological impact is of prime importance but other concerns, including the potential damages of the dam (earthquake, erosion, rupture of dam, etc.), should not be neglected;
- the change of the use of underground waters may be checked by means of constraints;
- radiological consequences within the long term must be assessed through the study of standard scenarios corresponding to possible hazards situations;
- the procedure has to be followed taking into consideration the concern of ensuring the diversity of the expertise and the correct and prompt information of the public;
- protection of the health of the population against the danger of ionizing radiation has to be in accordance with the EURATOM Directive no. 96/29. The dose limits applicable to the population have to be estimated in the most realistic way.

4.4. Disposal of sterile uranium ore and its management (some aspects)

Romania adopted the disposal of waste containing naturally occurring radioactive material (sterile uranium ore) in landfill sites. These sites are acceptable based on the fact that these radioactive materials are considered as non-radioactive on a radiation point of view following a radiological impact study carried out by qualified experts, in order to demonstrate that the annual effective dose received by the most exposed persons, due to the management of this waste, remains under the limit of 1mSv. A specific study have to be performed for each type of waste of each producer e.g. all CNU’s subsidiaries. The results have to be addressed to the administration of disposal sites along with the request for elimination as a basis for the definition of the source term to consider the impact study and to assess the acceptability of the wastes. When the dose is higher than 1mSv, the producer has to search an appropriate way to dispose of its waste taking into consideration the providing of the radiation protection programme.

5. Conclusion

The disposal of the mining processing waste (uranium ore) as well as the disposal of sterile in Romania is a very complex problem taking into consideration the importance and the need of the safety for such activities. The Romanian Nuclear Regulatory Body-CNCAN set up strict regulation and procedures according to recommendations of the IAEA and other international organizations. The National Uranium Company (CNU) has adopted and implemented adequate regulation and procedures in order to keep the safety of transport and the disposal site at international levels. The levels of the
estimated doses for transport and disposal are within acceptable limits provided by national and international regulations and recommendations.

REFERENCES

Prospects for disposal of low and intermediate level radioactive waste in Peru

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Peru

Abstract. Nuclear activities in Peru have been established for more than 30 years. These give rise to an amount of liquid and solid radioactive wastes and to spent sealed sources from industrial, medical and agricultural applications. This paper describes the waste management policy and strategy adopted by the Peruvian Institute of Nuclear Energy and the prospects for that policy, with special reference to the disposal of the low and intermediate level radioactive wastes. The repository considered is a concrete vault with backfill and containment barriers around the waste covered by a multi-layer cap. This system considers a drain control beneath the disposal vaults. The most important technical aspect considered for the near surface repository design is the engineered barriers because the site does not have good natural barriers.

1. Introduction
RACSO nuclear research centre operates a 10 MW pool type research reactor to produce different kind of radioisotopes. In addition, there is another facility for processing the radioisotopes, and many laboratories in which small research programmes are carried out.

The Nuclear Research Centre is located 40 km north of Lima and began operation in early 1989. It contains an area of about 15000 m² that will be used as a centralized waste processing and storage facility (WPSF). At this place are located a building for solid waste compaction, a trench for short half-life liquid radwaste, a small chemical precipitation plant and a room used for interim storage.

The waste inventory arising from the application of radioisotopes in medicine and industry basically comprises spent sealed sources. The operation of the Nuclear Research Centre has not produced a significant amount of waste when considered in relation to the total amount of waste produced by applications in industry and medicine.

The near surface repository must meet two basic objectives:

i) it must ensure the immediate and long term protective of the public and the environment;

ii) it must allow free use of the site, without any radiological limitations, after a maximum of 300 years [1].

A comprehensive strategy for the management of radioactive wastes considers the total waste inventory, waste treatment methods for the range of different waste types, time–scales for treatment for interim storage, waste acceptance criteria and the availability of ultimate disposal facilities. The developed waste management strategy helps avoid investments in poorly specified facilities and is likely to result in substantial cost savings. In the case of Peru engineered barriers have been considered as main consideration in the design of the repository because the site does not have good natural barriers.

2. Disposal system
The design of the bunker will take into consideration recommendations given by the IAEA [2]. It uses engineered barriers corresponding to the site conditions and a gravity catchment system to collect any runoff or infiltrating water entering during the operating periods. The bunker is small to facilitate the installation of engineered barriers and to guarantee its integrity in all circumstances. The waste packages, which are 0.2 m³ steel drums, are placed inside concrete disposal containers (Fig.1). Drums are then immobilized inside the container, forming a concrete block weighing 8 ton (Fig.2).
Containers are stored in disposal cells, each of which has a capacity for 42 containers and approximate external dimensions of 12.3 m x 7.7 m x 3.55 m (Fig.3). The containers are placed in contact with each other. Packages will be lowered by crane into the bunker in successive layers. After the last layer has been completed, the bunker will be completely filled with concrete and grout (Fig.3). The disposal cells are designed to withstand extreme loads, including site safety earthquake. Once the operation of each storage cell is completed, this is stuffed with gravel to stiffen the assembly.

FIG. 1: Dimension of concrete containers

FIG. 2: Dimension and distribution of cells

FIG. 3: Filling of the cells
3. Dimensions of the system for disposal

i) Concrete containers:
   - Length : 2,2 m
   - Width : 1,5 m
   - Height : 1,15
   - Wall thickness : 0,10 m
   - Capacity : 8 tons.

ii) Cells
   - Length : 12,3 m
   - Width : 7,7 m
   - Height : 3,5 m
   - Wall thickness : 0,40 m
   - Superior flagstone : 0,40 m
   - Inferior flagstone : 0,50 m
   - Surface : 94,5 m
   - Capacity : 42 containers/cell.

The distribution of the 42 containers for the cell is shown in Fig. 4. The capacity of drums, containers and cells is optimized to reduce costs in the case of developing countries. It permits to have a flexible system for their manipulation.

4. Conclusion

Geomorphology of the RACSO nuclear research centre does not appear to be suitable for a near surface repository, however that condition can be reverted considering engineered barriers in its design. In this case it is necessary to carry out an integrated safety assessment to demonstrate the stability and integrity of the near surface repository. IAEA has given technical recommendations in order to avoid that wastes lead to accidents, which could have negative impact on society.

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Radioactive waste management infrastructure in Madagascar


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Abstract. The law n° 97-041 on radiation protection and radioactive waste management in Madagascar was promulgated on 2nd January 1998 [1]. This law is based on the International Basic Safety Standards for Radiation Protection against Ionizing Radiation and for the Safety of Radiation Sources (BSS, IAEA Safety Series n° 115) [5]. This law was established by taking into account the recommendations of the IAEA experts. This law governs all activities related to the peaceful use of nuclear energy in Madagascar in order to protect the public, the environment and for the safety of radiation sources. Following the promulgation of the law, 4 decrees have been approved by the Malagasy Government. With an effective implementation of these decrees, the ANPSR will be the Highest Administrative Authority in the Field of Radiation Protection and Waste Management in Madagascar. This body is supported by an Executive Secretariat, assisted by the OTR for Radiation Protection and the OCGDR for Waste Management. According to the Decree n° 2002-569 on 4th July 2002, Madagascar-INSTN is the technical body of the Competent Authority in charge of the management of radioactive waste in Madagascar [2]. Inventory and advisory in waste management fields are among the most important tasks of the Radioactive Waste Management Department of Madagascar-INSTN. This paper gives an overview of the regulatory and technical framework of Radioactive Waste Management in the country.

1. Legislative and statutory framework

Law n° 97-041 on protection against the harmful effects of ionization radiation and radioactive waste management in Madagascar was promulgated on 2nd January 1998 [1]. This Law established the general legal framework within which all practices involving radiation risks must be carried out and regulated from the radiation protection and the radioactive waste management viewpoints. In its 59 articles, the Law defines the roles and functions of different organisations and introduces the fundamental radiation protection principles. This Law is considered reasonably comprehensive and generally consistent with the BSS requirements [5].

For the practical application of the Radiation Protection and Radioactive Waste Management Law (97-041), the INSTN issued four basic Decrees in 2002:

- (1) Decree n°2002-569 on 4th July 2002 dealing with designation, roles and functions of the" Autorité Nationale de Protection et de Sûreté Radiologique (ANPSR), Organe Technique de Radioprotection (OTR), Office Central de Gestion de Déchets radioactifs (OCGDR)",
- (2) Decree n°2002-1199 on 7th October 2002 dealing with the basic principles of radiation protection against ionizing radiation,
- (3) Decree n°2002-1274 on 16th October 2002 dealing with the basic principles of radioactive waste management,
- (4) Decree n° 2002-1161 on 9th October 2002 dealing with the detention and utilization of ionisation radiation sources in the medical field.

The first decree establishes the ANPSR as the regulatory authority. It establishes the OTR as the technical body for radiation protection and the OCGDR as the technical body for radioactive waste management. According to this decree, the INSTN will ensure the functions of the OTR and the OCGDR respectively. The decree became functional in 2004 when the regulatory activities were
transferred from Madagasacar-Institut National des Sciences et Techniques (INSTN) to the National Authority (ANPSR).

2. Responsibilities

2.1. Responsibilities of the regulatory body (ANPSR)

The ANPSR [2] is in charge:
- to define and to clarify the responsibilities of each concerned entity (Executive Secretariat, OTR, OCGDR, Users);
- to deliver, as the only one Competent Authority, authorization and licensing required by the law;
- to designate Inspectors, RP Officer, WM Officer on the OTR and OCGDR proposal;
- to propose draft legislation and regulation related to the RWM.

2.2. Responsibilities of the technical body in radioactive waste management (OCGDR)

The OCGDR [2] is in charge:
- to keep an inventory of all types of RW in Madagascar;
- to ensure the follow-up and the control of RWM at national level;
- to manage RW for which it is recognized that the generator is not able to manage the RW or has been deprived of authorization, or does not exist anymore or is unknown;
- to transmit relative specification of technical conditions and practical modalities for the treatment, conditioning, transport and disposal of radioactive wastes;
- to propose to the ANPSR all regulation concerning the treatment, discharge or elimination, management and the control of radioactive wastes and effluents;
- to carry out independent investigation in case of accident;
- to establish and to update emergency plan in case of accidents or emergency situation and to ensure the execution co-ordination of this plan;
- to organize training in RWM fields;
- to manage analysis and studies of interest to RWM;
- to ensure necessary contact between national and international organizations concerning the RWM;
- to co-ordinate intervention of local committees in case of a radiological emergency.
2.3. Responsibilities of the users

The first responsibility in safety [3] of RWM rests with the generators of the waste (producer, users). Each generator of RW holds to designate a person called « Waste Responsible » who is in charge of the management of the RW. The Waste Management Responsible has the following missions:

- to set-up and update inventory of radioactive sources and generated waste;
- to set-up contact and collaborate with workers which use radioactive sources;
- to ensure the link with the OCGDR;
- to establish data records system in order to facilitate identification, characterisation, collect and storage of radioactive source transformed into waste;
- to guarantee that the transport of RW on the site is in conformity with the written safety procedures;
- to ensure that before transportation, the transport procedures are in conformity with the specifications and acceptance of the OCGDR;
- to ensure the presence of appropriate shielding, labelling, physical protection and integrity of the packages;
- to ensure that wastes and effluents discharge must be done in conformity with the decontrol threshold and the authorisation;
- to report to the Direction all accidents or incidents related to the waste management in the installation;
- to preserve and to up-date all data concerning the drain off system of waste water of the site, municipal waste around the site, incinerator for non radioactive waste and all other installations related to the waste management.

The generator of waste must establish yearly an inventory of existing wastes and an estimation of waste generated by the installation in the future. This inventory should be transmitted to the OCGDR.

3. Budget

The expenditure of the ANPSR is composed particularly of the salary of the staff, the costs of intervention of the Technical Body (OTR and OCGDR) and other costs. The functioning budget of the ANPSR will be financed by the Government [2]. Interventions and allowance presentation services of the OCGDR will be remunerated. Intervention fees of the OCGDR are approved by the ANPSR [2].

Any person or institution which produces radioactive wastes is held to ensure their management and the related financial costs. [3]

4. Waste classification

Waste classification is mentioned in our regulation, particularly in the article nº 8 of the decree number 2002-1274 on 16th October 2002 [3], “Radioactive waste depending on the activity and half life of the radionuclide were classified as follows:

(a) Exempted waste;
(b) Low and Intermediate Level Waste (Short Lived);
(c) Low and Intermediate Level Waste (Long Lived);
(d) High Activity Waste.

5. Waste disposal options

For the future, the priority for Madagascar is the establishment of the Centralized Radioactive Waste Management Facility (for characterization and conditioning spent and orphan sources). A borehole for
SHARS (Sealed High Activity Radioactive Sources) and near surface disposal for LILW are the adequate site disposal for all radioactive wastes existing in Madagascar.

6. Conclusions

We are agreed that the matter required urgent action leading to the establishment of a centralized radioactive waste management facility. Such action could be initiated by the newly established Competent Authority (ANPSR), with the support of Madagascar-INSTN. For conditioning, particularly for neutron sources and SHARS (Sealed High Activity Radioactive Sources), IAEA’s assistance could be requested under RAF/03/005 Project. Madagascar makes efforts towards maintaining safety and security of radioactive sources, preventing theft, loss, damage, and unauthorized transfer of radioactive sources, by ensuring that:

- a periodic inventory of movable sources be conducted at appropriate intervals to confirm that they are in their assigned location and secured,
- a source must be managed and controlled within the legal regulatory framework,
- acquisition of radioactive sources with malevolent intent should be prevented,
- a source is not transferred unless the receiver holds a valid authorization,
- an emergency preparedness, regarding any decontrolled, lost, stolen or missing source is planned.

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Geological disposal of radioactive wastes in Russia

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Abstract. A huge amount of radioactive wastes is accumulated in Russia from nuclear weapons production, nuclear power plants operation and spent nuclear fuel reprocessing. As in other countries, an end point in the management of radioactive wastes is supposed to be their disposal in geological formations. In this paper a selective overview of this problem is presented with focus on geological conditions, providing for safe disposal of high-level nuclear waste.

1. Introduction.

The total radioactivity of radioactive wastes (RW) in Russia by the beginning of the year 2004 is about $2.3 \times 10^9$ Ci. Among them are:

- liquid high-level wastes (HLW) from nuclear weapons production accumulated at the Rosatom radiochemical combines (Production Association (PA) “Mayak” at the South Ural region, Siberian Chemical Combine (SCC) and Krasnoyarsk Mining and Chemical Combine (MCC) in Siberia);
- liquid (LW) and solid (SW) low-level and intermediate-level (IL) RW, vitrified and solid HLW from implementation of the closed nuclear fuel cycle, RW from research nuclear reactors operation and decommissioning of transport nuclear installations (PA “Mayak”, nuclear power plans (NPP), regions of nuclear fleet bases);
- LW and SW from radioactive sources used in medicine, scientific research and other fields. These last wastes are collected at the specialized combine “Radon”, which 16 enterprises are distributed over different parts of Russia.

The radioactive wastes (RW) management in the Russian Federation is regulated by a number of Federal Laws, Orders by the RF President, Government Orders and Regulatory Documents approved by other Federal Executive Authorities. The conceptual basis for the state policy in this field is summarized in the Federal Programme “Nuclear and Radiation Safety of Russia” for the period of 2000-2006 approved by the Government Order No. 149 of 22.02.2000. The essence of this policy is formulated as “creation of the pilot-industrial facilities for RW disposal at the North-West, PA “Mayak” and Krasnoyarsk MCC regions, as well as at the Far East region”.

In accordance with the regulatory documents being in force in Russia and conventional practice, RW are conditioned, that is, are transformed to forms adequate for their safe confinement, with the prospect of their subsequent disposal with observance of necessary safety requirements. Efficient technologies for conditioning and disposal in sub-surface and on-surface facilities of the short-lived low-level and intermediate-level RW which permit to secure reliable isolation of this wastes during first hundreds of years are successfully utilized in Russia and abroad. As will be shown below, the problem of safe isolation of HLW is much more complicated.

2. Underground disposal of liquid low-level and intermediate-level wastes

Russia practices underground disposal of low-level and intermediate-level liquid radioactive wastes. LW are injected into the deep-seated permeable sandstone horizons embedded in the layered sedimentary sequence of artesian basins with the low-rate groundwater movement or stagnant groundwater regime. Radioecological safety of the injection sites is secured by: their location at the
seismically stable regions; presence of the low-permeable clay horizons above and below of the injection horizon; presence in geological section above of the injection horizon of one or more that one buffering water-bearing horizons with potential capacity to absorb contaminants in case of their upward migration; presence in the injection horizon of clayey and mixed-layered aluminosilicate minerals with high sorption capacity; low (not more than first meters per year) groundwater flow velocity in the injection horizon; remote position of the injection sites from the discharge zones, providing for groundwater travel time exceeding the time of preservation of the disposed radionuclides biological toxicity.

Deep well injection is practiced in Russia since the middle of 60th. With use of this disposal option were reliably isolated from the ecosphere LW with the total volume of several tens of millions of m³. At present, deep well LW injection is executed at the three injection sites: Scientific Research Institute of Nuclear Reactors (NIIAR, Dimitrovgrad), SCC (Tomsk-7), and MCC (Krasnoyarsk-26). The experience obtained has shown that geological, hydro-geological and hydro-geochemical characteristics of these injection sites, with provision of observance of the all technological regimes of sites exploitation, reliably secure isolation of the injected low-level and intermediate-level LW over the period of their radiobiological toxicity.

3. Underground disposal of HLW

Because of the complexity and multifactor nature of the problem of HLW isolation, Russia, as other countries, follows a multibarrier strategy for isolation of HLW repositories, which implies combination of engineering and natural barriers precluding radionuclides escape into the inhabited environment.

3.1. Engineered barriers

In resolving this problem, the main efforts of Russian researchers are focused on the development of matrix materials for solidification of liquid HLW that are appropriate for the immobilized HLW long-term storage and disposal.

Waste forms for liquid HLW immobilization

In Russia liquid HLW derived from spent nuclear fuel reprocessing are incorporated at the RT-1 plant, PA “Mayak”, into the Na-Al-P glass-like matrices. This glass composition is easily devitrified and has a relatively high solubility in underground waters. Nevertheless, solidification of liquid HLW is in any case justified, as storage of solid HLW is much less dangerous than that of liquid one.

Crystalline matrices produced from mixture of Ti, Zr, Al, Ca, Fe, Si, P oxides are much more stable. Radionuclides enter the structure of these phases due to isomorphic exchanges with major elements. Scientists from the Russian Academy of Sciences in cooperation with specialists from Institutions of Rosatom RF and Moscow State University have developed and studied promising matrices composed of titanate with fluorite structure (pyrochlore, murataite); silicates with apatite structure, alumoferrites with garnet-type lattice and some other.

Studies are on-going for elaboration of matrices with high isomorphic capacity in respect to tetravalent actinides and other elements of HLW streams, durable at conditions of radiation and underground water attack. Among such universal matrices titanate murataite-based forms and ferrite garnet ceramics are most promising for immobilization of actinide-containing radioactive waste with complex composition.

3.2. Geological medium as a main isolation barrier of HLW repositories

The geological medium starts to play a role of the main isolation barrier in prevention of radionuclides escaping from the HLW repository to the ecosphere since the time of deterioration of the engineered barriers, what is supposed to take place after the first thousand years after repository closure.

As appropriate geological media for HLW disposal are considered clays, salts, or different crystalline rocks. In Russia, crystalline rocks are considered as preferred type of geological media.
3.2.1. Uranium ore deposits as natural analogues of HLW repositories

The main argument in favour of the conception of uranium ore deposits as natural analogues of HLW repositories comes from that pitchblende, the main ore mineral of these deposits, is the full analogue of synthetic uranium dioxide – uraninite, – which comprises more than 95% of the spent nuclear fuel (SNF), in reprocessing of which arises HLW. It is of prime importance that in nature exist geological situations providing for isolation from the ecosphere of many uranium ore deposits (or their parts) over periods of millions of years since the deposit formation. This observation witnesses to the possibility in principle of the HLW safe underground disposal in geological media.

Of special interest, as natural analogues of future HLW repositories, are uranium deposits in which are present ore aggregations with the fully preserved primary uranium mineralization, and, along with this, are present manifestations of the post-ore process that caused mechanical and mineral-chemical destruction of the primary uranium ore bodies.

Such contrasting combinations of geological-structural and mineral-geochemical factors, which provided, on the one hand, for preservation of the integrity of primary uranium ores, and, on the other hand, for formation of their altered varieties, were studied by authors in detail at the unique, by their uranium reserves, deposits of the Streltsovka ore field (Eastern Transbaikalia region, Russia). These studies were carried out under the Project No. 1326 of the International Science and Technology Center (ISTC) in Moscow funded by the Radioactive Waste Management and Funding Center (RWMC) of Japan.

Streltsovka ore field includes 19 uranium ore deposits located in volcanic caldera of Mesozoic age with the area of about 150 km² and partly in Paleozoic granites of the caldera basement. Pitchblende, the main uranium ore mineral of the deposits, was subjected to a different extent to the overprint by the endogenous post-ore hydrothermal solutions what caused its partial pseudomorphous replacement by the Si-U gel. It was established that uranium liberated in this process was re-deposited as the same Si-U phase in the close vicinity to the apopitchblende pseudomorphs. The practically full immobilization of liberated uranium by the newly formed gel phase was confirmed quantitatively.

The data obtained permit to expect that the impact of hydrothermal solutions would not cause any significant removal of uranium from the underground HLW repository.

3.2.2. Selection of geological media and candidate sites for HLW disposal by the example of the PA “Mayak” region

Production Association “Mayak” is located at the eastern slope of the Ural range in the Chelyabinsk oblast. The enterprise was created in 1948. Till 1986 it produced weapons plutonium, and since 1976 executes nuclear fuel reprocessing (plant RT-1). Liquid HLW from previous activities and currently generated are solidified in the glass-like aluminophosphate matrices. The amount of vitrified HLW is above 3000 t with the total radioactivity up to $500 \times 10^6$ Ci.

For underground isolation of vitrified HLW within the PA “Mayak” region the most appropriate host medium is the sequence of volcanogenic rocks of basic composition about 2 km in total thickness. This sequence in composed by the andesite-basalt porphyrites and their tuffs, tuff-lavas and tuff-sandstones with the low effective porosity (~ 0.2-0.3%), high mechanical strength and thermal stability. As a whole, the sequence is characterized by relatively uniform chemical composition. By the data from experimental studies, volcanites porosity remains unchanged at elevated temperature (up to 200°C) and pressure. In presence of hot waters, the primary minerals of the volcanite sequence (Ca-Na plagioclase, pyroxene, olivine) are replaced by secondary minerals (epidote, chlorite, hydromica, mixed layered and clayey minerals, Fe and Mn hydroxides) which have high sorption capacity with respect to radionuclides. Besides that, secondary minerals, because of their higher specific volume, “seal” rock pores and microfractures. The above mentioned volcanites properties permit to characterize them as a suitable host for HLW disposal.

At the same time, the PA “Mayak” territory is characterized by the intense but nonuniform tectonic disturbance. Regional fault zones of the different strike and thickness are present. Within these zones rocks are subjected to intense schist formation, crushing, and are dissected by tectonic seams with displacements of the post-Quaternary time. These zones are characterized by high permeability and are...
the main groundwater conduits in the rock basement. Taking into account high hydraulic conductivity of these zones and the potential chance of their “rejuvenation” by the new tectonic movements, the areas occupied by these fault zones are considered as unsuitable for siting engineered constructions of disposal facilities.

The geo-blocks between the fault zones are much less tectonically disturbed. In such blocks, located within the PA “Mayak” sanitary protection zone, were chosen two areas recommended for the more detailed studies with the objective of site selection for construction of the underground research laboratory (URL) and later on – of the HLW repository.

Taking into account the relatively limited spatial dimensions of the chosen areas (2 and 4 km²), it seems obvious that they are not suitable for construction of the mined repository. As the more realistic option seems to be the design of the URL through drilling large diameter wells (not less than 600 mm), the practice that is well developed in Russia, with the subsequent construction of the deep-well repository for vitrified HLW.

From the data on geology of the PA “Mayak” sanitary protection zone territory and the preliminary results obtained from computer simulation of hydrodynamic conditions of HLW disposal, as an optimal depth of the future repository location is supposed to be the depth interval from 500 to 1000 m from the day surface.

4. Geological basis for the long-term safety evaluation of HLW disposal

In the normative documents regulating activities in the field of HLW disposal in Russian Federation, a principle of the stepwise assessment of the implemented works is formulated. The decision on transition to the next stage of work depends on safety assessment of the results obtained at the previous stage. The final decision on HLW disposal could be approved only after going through the all necessary procedures of safety evaluation.

As conclusive criteria for safety evaluation of the future HLW repositories, in the Russian Federation, as in other countries, are accepted levels of the potential radionuclides escape to the ecosphere transformed to evaluation of cancer risk (10^-5-10^-6 per year) and individual radiation dose for population (0.1-1.0 mSv/year) calculated over a period of 10 000 years. Computation of such safety indexes is a task to be solved in future.

At the early stages of work, in Russia for safety evaluation are used geological criteria for examination of the principal suitability of geological medium for safe HLW disposal and determination of the possibility to create reliable model of radionuclides migration to the biosphere.

At the site selection stage, criteria for evaluation of geological media suitability include requirements imposed on characteristics of host rocks, groundwaters, tectono-dynamic conditions and presence of mineral reserves.

At the stage of site characterization, the objective of work is in final selection of the site of a sufficient area with suitable geological structure. Along with the data obtained from the integrated geological-geophysical and engineering-geological studies, the scales and uncertainty limits are determined for the long-range scenarios of the possible changes of geological situation and conditions. These data are used for the adequacy evaluation of the future model of the possible radionuclides escape to the biosphere.

Geological uncertainties at the stage of site characterization come mainly from the deficiency of the geological-structural data. With the objective of elimination of these uncertainties or maximum possible reduction of their influence on safety evaluation, a complex of the following specialized studies is carried out:

- analysis of the stress-strain state of geological medium at present and in the past tectonic periods;
- comparative analysis of the contemporary and the past geodynamic regimes in the studied geological blocks for the long-term prediction of the future state of geological medium;
experimental studies of the effect exerted by tectonic, thermal and radiation loads on the rock properties of candidate geological blocks.

For collection of an additional data, necessary for substantiation of a decision on safety assessment of future HLW repositories, specialized studies should be carried out in the underground research laboratories, which construction and operation is an integral part of the work program preceding the stage of taking a final decision on creation of the HLW repository.

5. Long-term safety prediction of HLW disposal with use of mathematical modelling techniques

As the main mechanism of radionuclides escape from a repository of radioactive waste to the biosphere is their transport by groundwater. The forecasts of radionuclides migration in the underground medium, carried out in Russian Academy of Sciences, are based on mathematical models of radionuclides transport by groundwater taking into account convection, diffusion, dispersion mechanisms and redistribution of radionuclides between the liquid and solid phases. A necessary condition for determination of these forecasts is calculation of groundwater flow velocity fields. Radioactive decay processes in the waste can lead to a significant heat generation. As a result of a corresponding increase of a temperature in the disposal zone, the groundwater flow in the disposal zone can have both forced component (caused by the regional flow of the groundwater) and free thermal component. The last depends on temperature distribution in the disposal zone. Therefore, determination of the radionuclides migration forecast calls for solution of the coupled problem of groundwater flow, heat and mass transfer in the underground medium. A computer simulation of radionuclides migration will be carried out for three cases: (1) potential underground repositories (of mined and deep-well types) for disposal of solidified HLW and SNF, (2) sites of deep-well injection disposal of liquid radioactive waste, and (3) disposal facilities of radioactive materials created in early years of nuclear studies carried out close to large cities. In the models will be taken into account complex structure and heterogeneity of the underground medium where radionuclides migrate, presence of large-scale tectonic discontinuities. Significant attention is paid to preparation of input data for the modelling (including development of original techniques).
Long-term behaviour of engineered barriers used in surface disposal

1. Safety functions and treatment of uncertainty

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Abstract. Repository safety assessment (SA) requires defensible assumptions about the future state of engineered barriers (EB). In this study, a reference scenario is presented that describes the gradual EB degradation by means of a step function. To address uncertainties about the assumptions of the future EB state used in the reference scenario, “what-if” scenarios were developed. The analysis of long-term EB performance also included calculations with two different conceptual models characterized by homogeneous or heterogeneous flow. Finally, consequence analysis were done with best estimate parameter values but also with variations of parameters according to specific distribution functions.

1. Introduction

NIRAS/ONDRAF is co-ordinating the Belgian disposal programme of low- and intermediate-level short-lived radioactive waste (LILW-SL). Within this programme, one considered option is near-surface disposal. This paper discusses the assumptions used in the modelling of engineered barrier (EB) performance for a proposed surface repository. One of the most important questions that we try to address here is "Until what time in the distant future can the near field models be applied in the estimation of radionuclide release as a basis for dose estimation?" The answer to this question is needed to be confident that the future impact is not underestimated. An accompanying paper discusses the degradation mechanisms accounted for in SA [1].

2. Expected performance of engineered barriers for near-surface disposal

2.1. Components of the engineered barriers within a near-surface repository

The Belgian near-surface multi-barrier repository concept encompasses six components: (1) waste form, (2) waste drum, (3) monolith filling grout, (4) monolith walls, (5) module, and (6) multi-layer soil cover (FIG. 1). The expected performance of the EB differs between the institutional control period (ICP), i.e. 200 to 300 years after waste emplacement and construction of the cover, and the long-term period. The latter starts after the ICP and continues until the considered assessment time cut-off (varying from 10 000 years up to 100 000 years or more; no time cut-off has yet been specified by the Belgian safety authorities).

The expected EB performance is expressed in terms of safety functions (Table 1). Three safety functions are considered:

- Limitation of access (L): direct access to the waste is hindered, and direct radiation is sufficiently isolated from man and environment;
- Limitation of water ingress (C): because water is the most important vector for radionuclide spreading, contact between water and radionuclides is being limited as much as possible;
- Diffusion and retention (R): processes such as molecular diffusion, sorption and precipitation all result in a retarded and spread radionuclide release, and thus contribute to a significant reduction in the radionuclide flux being released from the repository.

The expected EB performance for the long-term period corresponds to the initial performance at $t = 0$. Various degradation mechanisms will lead to gradual degradation of the EB and thus result in a gradual decrease in the fulfilment of the safety functions attributed to the EB components.
Table I. Expected performance of EB for near surface disposal in terms of safety functions

<table>
<thead>
<tr>
<th>Component</th>
<th>Expected safety functions during ICP (200 to 300 year)</th>
<th>(Initially) Expected safety functions for long-term period starting after ICP</th>
</tr>
</thead>
<tbody>
<tr>
<td>Waste form</td>
<td>R is a safety reserve</td>
<td>R is safety reserve</td>
</tr>
<tr>
<td>Waste drum</td>
<td>C and R are safety reserves</td>
<td>C and R are safety reserve</td>
</tr>
<tr>
<td>Monolith filling grout</td>
<td>C, R, L</td>
<td>C, R, L</td>
</tr>
<tr>
<td>Monolith walls</td>
<td>C, R, L</td>
<td>C, R, L</td>
</tr>
<tr>
<td>Module</td>
<td>C, L (R is safety reserve)</td>
<td>C, R, L</td>
</tr>
<tr>
<td>Cover</td>
<td>C, L</td>
<td>L (C is safety reserve)</td>
</tr>
</tbody>
</table>

3. Treatment of uncertainties in assessing the performance of engineered barriers

Several uncertainties exist about EB evolution and degradation. Within SA key uncertainties are generally addressed using scenario variants, conceptual model variants, and parameter variations. For modelling EB evolution throughout the ICP and the long-term period, distinction is made between the cover and the concrete components (module and monoliths). The safety functions can be attributed to these EB, but not all safety functions will be fulfilled completely throughout the entire evaluation period owing to EB degradation. Assumptions regarding cover and concrete degradation are further discussed.

![FIG. 1: Multi-barrier repository with drums & monoliths (1), module (2), soil cover (3), and drainage gallery (4; courtesy MONA). Degradation states considered in reference scenario (B).](image)

4. Cover degradation

Cover degradation will result in a loss of the safety function C “limitation of water ingress”, whereas the safety function L “limitation of access” is still maintained. We conservatively put the cover lifetime at 300 y in the reference scenario, which corresponds with the ICP. During this period, degradation of the cap can be repaired. On the basis of unsaturated flow calculations, we demonstrated that the multi-layer cap diverts 99% of the infiltrating rainwater, resulting in a water flux at the bottom of the cover equal to $\sim 10^{-10}$ m s$^{-1}$ [2]. After the ICP, presumed degradation of the multi-layer cap results in an increase in infiltration beneath the cover (from $10^{-10}$ to the net rainfall of $8.6 \times 10^{-9}$ m/s).

5. Module and monolith evolution and degradation

5.1. Introduction

During the ICP, concrete EB are intact and fulfil the safety functions L, C, and R. As a result, only very little infiltration water enters the module from the top through an intact roof (equal to the rate of water percolating across the bottom of the overlying soil cover, i.e. $10^{-10}$ m s$^{-1}$, because the saturated hydraulic conductivity $K_s$ of the roof is supposed to be higher than $10^{-10}$ m s$^{-1}$). The infiltrating water is further percolating between the monoliths, and is collected at the bottom of the module and diverted through a series of drainage pipes into the inspection gallery. During the ICP there is no release of...
radionuclides via the water pathway. The numerical flow and transport model implemented for this period used only best estimate values for physical and chemical parameters that govern radionuclide migration [3]. The best estimate values are assumed representative for intact engineered barriers.

After the ICP, degradation of the concrete components will initially affect the safety function C. However, as time and thus degradation progresses, also safety function D can be affected. Since proper knowledge on how to describe barrier degradation is currently missing, considerable uncertainty exists about how to treat EB evolution in SA. The uncertainty has been tackled here by invoking variant scenarios in addition to a reference scenario, by comparing different conceptual models, and by using different parameter values in the consequence analysis.

5.2. Scenarios considered

Uncertainty about the EB future states was addressed by developing so-called variant scenarios, in addition to one reference scenario. The variant scenarios are “what-if” scenarios, and are based on either very optimistic or very pessimistic assumptions. Three such variant scenarios were evaluated: a normal evolution scenario (NES), and two altered evolution scenarios (AES-I and AES-II). The NES assumes all barriers remain intact throughout the entire evaluation period. AES-I assumes modules can no longer fulfill safety function C, and monoliths can only partially safety function C, but safety function R is still operational. AES-I occurs after repository closure. In the most pessimistic scenario (AES-II) safety function C is completely lost for modules and monoliths immediately after repository closure. As was true for AES-I, safety function R is assumed to be still active.

A reference scenario was further defined in which the continuous degradation processes and the loss of repository performance was modelled as a step function. Only natural causes of degradation were accounted for. Four major periods were distinguished corresponding to a different state of the engineered barriers (FIG. 1). In the first period, which covered the ICP, all barriers function as required and no degradation is considered (state 1-1-1). The second period, from 300 to 10 000 years after closure, assumes the soil cover would be degraded soon after the ICP, while the concrete barriers maintain their initial characteristics (state 0-1-1). A third period considers complete degradation of modules and partial degradation of monoliths between 10 000 and 100 000 years (state 0-0-1). The last period, for times in excess of 100 000 years, assumes all concrete barriers have lost their capacity to reduce infiltration of rainwater (state 0-0-0), although the pH is still relatively high to guarantee sorption and hence spread and retarded release. The underlying assumptions regarding degradation of the concrete engineered barriers are discussed below.

**Period between 300 and 10 000 years.** Short-term concrete degradation may occur due to corrosion of reinforcement bars. As a result, module components will likely produce micro-fissures and cracks resulting in localised increased infiltration. Long-term concrete degradation is likely to be determined by the rate at which Ca(OH)$_2$ and subsequently the hydrated calcium-silicate (CSH) phases will be leached from the concrete. The process of calcium leaching results in a pH drop and an increase in porosity and hence an increase in hydraulic conductivity. Several studies suggest that the first 10 000 years pH in concrete monoliths would be sufficiently high (i.e. above 11) to maintain a high sorption and low solubility, but also to assume limited changes in porosity and hydraulic conductivity (e.g. [4]). Therefore, we assume the near field model for modules and monoliths used for the first 300 years can be used up to approximately 10 000 year. We further assume the roof has no resistance anymore towards the infiltrating water. Monoliths are still intact (with their initial hydraulic conductivity $K_s$ of $3 \times 10^{-12}$ m/s). Because the internal drainage system has been inactivated, contaminated water from within the module will penetrate slowly into the concrete floor through which it will leave the repository. Because the floor is considered to be still intact, the flow rate will be determined by its saturated hydraulic conductivity, equal to $3 \times 10^{-10}$ m/s. Water will then flow through an unsaturated zone until it reaches the groundwater. As a result, release of radionuclides to the environment occurs.

**Period between 10 000 and 100 000 years.** We assume that after 10 000 years, concrete modules (walls and floor) have been completely degraded (i.e. rainwater enters the module at a rate equal to the net rainfall rate), whereas monoliths have also undergone an initial phase of degradation resulting in a hundred-fold increase in conductivity. Several FEPs may have contributed to this degradation, including corrosion of reinforcement structures, leaching of Ca(OH)$_2$ resulting in loss of mechanical strength, and changes in physical properties such as hydraulic conductivity owing to pH decrease.
**Period beyond 100 000 years.** For times beyond 100 000 y, we assumed all concrete EB would have completely degraded such that infiltrating rainwater can have a maximum access to the waste. However, since the quantity of cement used will be huge, pH will be strongly buffered. On the basis of calculations on the pH evolution performed by [4] initial estimates suggest that pH would be > 10 for at least one million y. In other words, although the physical degradation of concrete may have evolved to its final stage, the chemical degradation is more attenuated owing to the large quantities of cement. As a result, geochemical conditions will still be favourable for high sorption and low solubility.

5.3. **Conceptual models considered**

The disposal concept considers rows of monoliths being stacked one on top of the other inside a module, up to six monoliths high. Voids in between monoliths are not filled. The latter induces uncertainty about the water flow regime inside the module, i.e. if void spaces become gradually filled by natural cementation processes preferential flow paths in between monoliths may not be present. If, on the other hand, void spaces remain unfilled throughout the entire evaluation period, then such flow paths will exist and a different flow, and hence radionuclide migration pattern may exist too. Therefore, two alternative conceptual models were developed: one that considers flow through a homogenous concrete block composed of monoliths without water bearing voids, and the other with several channel-like water bearing voids.

5.4. **Parameter values considered**

Uncertainty about long-term EB performance was also addressed by allowing parameters to vary around best estimate values, according to distribution functions defined in [3]. Parameters considered mainly had an effect on the safety function R. In this way calculations were done with parameters that reflected a poor chemical state of the EB, i.e. with low sorption and high diffusion.

6. **Conclusion**

Long-term evolution and degradation of EB in SA has been addressed by (1) defining and evaluating a reference scenario that distinguishes different timeframes characterized by a stepwise increase in EB degradation, (2) developing “what-if” scenarios that provide an envelope around the reference scenario (i.e. scenarios that are more optimistic or pessimistic than the reference), (3) building contrasting flow and transport models, and (4) including parameter variations in consequence analysis.

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Long-term behaviour of engineered barriers used in surface disposal

2. Degradation mechanisms accounted for in safety assessment

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Abstract. Repository performance assessment requires defensible assumptions about the future state of engineered barriers (EB). Because of the overwhelming complexity of degradation mechanisms that affect the long-term performance of EB, simplified concepts have to be built in conceptual and numerical models. In this paper we discuss how such degradation processes were simplified and conceptualised into a series of models used in safety assessment (SA).

1. Introduction

Within the Belgian disposal programme of low- and intermediate-level short-lived radioactive waste (LILW-SL), near-surface disposal is one considered option. In an accompanying paper \cite{1}, the near-surface disposal concept and the SA methodology are explained. This paper discusses (1) the simplifications assumed in modelling long-term degradation of concrete EB, (2) the scenarios used to represent degradation, (3) the conceptual and numerical models of flow and transport, and (4) associated parameter values.

2. Safety functions to be fulfilled by concrete components of engineered barriers

The Belgian near-surface multi-barrier repository concept was described by \cite{1}. The performance of the different EB is expressed in terms of safety functions \cite{1}:

- Limitation of access (L): direct access to the waste is delayed, and direct radiation is sufficiently isolated from man and environment. For the long-term (i.e. $t >> 300$ y), L is assured by monolith filling grout, monolith walls made from concrete, concrete module, and cover.

- Limitation of water ingress (C): as water is the most important vector for radionuclide spreading, contact between water and radionuclides is being limited as much as possible. Long-term limitation of water ingress is guaranteed by monolith filling grout, monolith walls, and module.

- Diffusion and retention (R): processes such as molecular diffusion, sorption and precipitation all result in a retarded and spread radionuclide release, and thus contribute to a significant reduction in the radionuclide flux being released from the repository. This safety function is secured by monolith filling grout, monolith walls, and module for the long-term period.

Various degradation mechanisms during the long-term period will lead to gradual degradation of the engineered barriers and thus result in a gradual decrease in the fulfilment of the safety functions attributed to the different components of the EB. How such degradation processes were accounted for in the present SA will be further discussed.

3. Description of processes governing concrete EB degradation

The initial state of the concrete EB is characterized by particular physical, chemical, and mechanical properties that secure fulfilment of the safety functions for at least the duration of the institutional control period (ICP), and that are in line with the design requirements formulated by IAEA \cite{2}. Concrete monoliths envisaged in the Belgian concept will have very low hydraulic conductivities ($< 10^{-11}$ m/s), thus ensuring limitation of water ingress. Initial pH will be high (between 13-14), and guarantees ideal conditions for high retention by sorption and low solubility. In addition to pH, cement composition may also contribute to the immobilization potential owing to its low internal redox
potential. Mechanical (i.e. compressive) strength of module and monoliths will be sufficiently high to provide the necessary mechanical resistance and thus provide stability to the disposal modules and the overlying soil cover.

Most important degradation mechanisms affecting the lifetime of concrete EB are physical and chemical degradation. Physical degradation is defined here as related to modifications of physical properties of a porous medium (hydraulic conductivity, porosity, bulk density, among others), whereas chemical degradation refers to changes in chemical properties (solubility, sorption, carbonation, etc.) that affect, directly or indirectly, the migration potential within an EB. The end of the EB lifetime can be due to two major features: (1) increase in hydraulic conductivity resulting in flowing water getting into contact with the waste, (2) loss of load-bearing capacity. Major mechanisms contributing to these conditions are degradation processes and cracking processes.

Degradation processes include sulphate attack, corrosion of reinforcement steel, alkali-aggregate reactions, and leaching by infiltrating soil water. Naturally occurring sulphates may degrade concrete by causing decalcification of the hydrated calcium-silicate (CSH) phases, which is the major binding component of hydrated portland cement. Corrosion of reinforcement steel is generally caused by chloride ions and by carbonation. Steel corrosion results in corrosion products being formed, which results in stress increase followed by concrete cracking if the tensile strength is exceeded. Aggregates present in concrete react with alkali species, which sometimes results in formation of expansive products causing cracking. This cracking can significantly increase hydraulic conductivity. Finally, water percolating through concrete dissolves soluble materials, making the material more porous and hence also affects flow and transport properties in a negative way.

Cracking processes include load-induced cracks, drying shrinkage, cracks due to thermal and moisture expansion/contraction, freezing and thawing. Structural cracks are induced by loads, foundation shifting, and seismic events. A second type of cracks (intrinsic cracks) occurs as a result of internal stresses within the concrete owing to volumetric expansion or shrinkage.

Changes in the physical properties (mainly hydraulic conductivity, i.e. safety function C) of concrete owing to cracking and degradation are likely to occur in the first few hundred to thousand years, especially for large concrete structures such as the module. Changes in chemical properties determining the immobilization (safety function R) are generally considered to progress at a much slower rate (tens of thousands of years may be required to bring the initial pH below 11) owing to the large amount of cement. As a result, the chemical barrier (safety function R) will most likely still be operational at very long times, even though the physical barrier has already been completely degraded.

In the current SA, individual degradation processes as such were not modelled. Rather was the overall effect of degradation and cracking on hydraulic conductivity increase accounted for. Other physical properties of concrete such as porosity and bulk density remained unchanged during the evaluation period. Stochastic sensitivity analysis indicated that for most radionuclides considered variations in porosity and bulk density were only minimally affecting to model output. Hence this provides a first justification for the assumption of treating those properties as constant. Effects of degradation on chemical properties such as solubility and sorption were not considered directly in none of the scenarios.

The effect of changes in sorption parameter was indirectly evaluated by means of stochastic sensitivity and uncertainty analysis. A more rigorous treatment of sorption and solubility as function of degradation stage is recommended in further studies. For this purpose, separate geochemical calculations are recommended for quantifying the pH-dependent sorption and solubility.

4. How can degradation processes be simplified and conceptualised in safety assessment

Representation of individual degradation processes and their interactions in a SA model considering time scales of tens to hundreds of thousands of years, is currently not feasible. Although several ‘service-life’ models are available that describe degradation by means of simplified empirical models, their application is often limited to ‘short-term’ evaluation (up to 500 - 1000 y). Furthermore, the more
mechanistically oriented models either emphasize on chemical or physical degradation. Coupling between physical-chemical-mechanical processes is an ongoing field of research [3].

A rather pragmatic approach was followed in the present study. First of all, degradation of concrete EB was assumed to only affect the hydraulic conductivity, while other physical and chemical characteristics remained unchanged. A stepwise increase in hydraulic conductivity was imposed for modules and monoliths, which was then implemented in the reference scenario. Such increase in hydraulic conductivity was assumed to occur everywhere and at the same time for a particular EB component. Thus crack formation and the associated local increase in hydraulic conductivity was not modelled as such. A single-porosity medium with uniform hydraulic properties was assumed.

A 2D finite element variably-saturated flow model was run until steady-state flow was achieved. Two alternative models were developed: the first considered flow through a homogeneous block composed of monoliths without water bearing voids in between adjacent monoliths, the second had several channel-like water bearing voids in between monoliths. Based on the steady-state water velocity profile, transport calculations were made. A 1D transport model was used in combination with the homogeneous flow model (since flow was vertically uniform), whereas a 2D model was used to account for flow and the concomitant transport heterogeneity (i.e. low flow inside monolith, relatively high flows within void spaces). The latter model was only applied to the period 300-10 000 y.

Mathematical models accounting for a quasi continuous change in material properties can be easily implemented in future studies, but their use is defensible only if sufficient information is obtained about the expected evolution of material characteristics as degradation progresses. Future studies should therefore emphasize on collecting the necessary information on long-term (at least up to 10 000 y) material behaviour with which a practical model can be build for use in routine SA calculations.

5. Scenarios used for modelling degradation

Scenarios used as a basis for SA modelling are detailed in [1]. A reference scenario was defined in which the continuous degradation processes and the loss of repository performance was modelled as a step function. Each step corresponds with one or several EB being no longer able to fulfil safety function C. Furthermore, uncertainty about the EB future states was addressed by developing so-called “what-if” scenarios. Two of these scenarios also consider EB degradation, and the type and degree of degradation is also present in the reference scenario, but the timing is different, i.e. degradation is assumed to occur after repository closure. These are pessimistic scenarios. One optimistic scenario considered intactness of all EB for the entire evaluation period. Thus, we mainly addressed the uncertainty about when degradation would occur, i.e. after closure, 10 000, or 100 000 y?

As mentioned earlier, the only degradation accounted for is the loss of safety function C. By considering both optimistic and pessimistic scenarios, envelopes around the real but unknown behaviour have been established. More pessimistic scenarios than AES-II (see [1]) or more optimistic scenarios than the NES (see [1]) are not likely to be found, at least not in terms of safety function C. Future work will therefore be oriented towards a refinement of the reference scenario. First of all, the step function should be refined on the basis of defensible relationships between stage of degradation (i.e. time) and physical properties. Needless to say that such relationships need to be determined on the basis of local boundary and site conditions and concrete composition specific for the investigated repository concept. It also means allowing other material properties such as porosity, density and diffusion coefficient to be updated as function of stage of concrete degradation.

Existing scenarios can be further refined to also explicitly include an evaluation of the safety function R (diffusion and retention). This could include development of a scenario in which chemical degradation (i.e., updating of sorption and solubility parameters for different pH values) is considered as a step or continuous function. Additional pessimistic scenarios may be developed to obtain envelopes around some reference scenario in which effects on safety function R are evaluated.

6. Parameter values used in SA modelling

Numerical models of flow and transport require a considerable amount of model parameters. Selection of model parameters was based on a formal QA procedure referred to as ‘Data Collection Forms (DCF)’. The DCFs provide a unique link between SA modelling and parameters, and are the basis for
parameter representativeness and traceability. Basic data used in derivation of best estimate parameter values is provided in the DCFs [4]. Also provided are statistical distribution and associated statistical parameters for use in stochastic calculations. Since most of the basic data was obtained from a literature survey, several screening criteria were applied to identify which data was representative for our EB. For example, selection of appropriate hydraulic conductivity values for high-quality concrete was based on a water/cement ratio in the range 0.4-0.5. The resulting $K$ database had values between $1.9 \times 10^{-13}$ and $3.9 \times 10^{-11}$ m/s, with a best estimate of $2.75 \times 10^{-12}$ m/s (rounded to $3 \times 10^{-12}$ m/s). The latter value was assigned to the monoliths as initial value. Concrete walls and floor from the module were given a 100× higher $K$ value, since large concrete structures are more susceptible to crack formation than small containers. For the concrete roof a $K$ value of $3 \times 10^{-9}$ m/s was assumed in the reference scenario. Thus no barrier function is attributed to the roof in this study, as a conservative assumption.

In the models used to evaluate the degradation of concrete EB, only $K$ was changed. For modules, the roof was degraded first, i.e. from $t \geq 300$ years onwards (weak degradation in [1], Fig. 1). Its $K$ value increased to a value large enough to allow all infiltration water to enter the module. Since the imposed water flux from a degraded cover was $8.6 \times 10^{-9}$ m/s, $K$ for the roof was put at $3 \times 10^{-7}$ m/s. The next degradation (strong degradation in [1], Fig. 1) occurred between 10 000 and 100 000 y. Within this period, module walls and floors degrade completely; i.e. there is no resistance to infiltrating water. Degradation has now also affected the monoliths, with an assumed increase in $K$ from $3 \times 10^{-12}$ to $3 \times 10^{-10}$ m/s. The monoliths thus still fulfil the safety function C, but to a limited extent. The last degradation step (very strong degradation in [1], Fig. 1) occurs at $t \geq 100 000$ y, when $K$ for monoliths increases again to its final value (large enough to accommodate a water flux of $8.6 \times 10^{-9}$ m/s).

Future work will address the flow and transport parameters that were shown, on the basis of a stochastic sensitivity analysis and on uncertainty analyses, to have the highest impact on the calculated dose: (1) initial $K$ value for monoliths, which means performing $K$-measurements on core samples taken from prototype monoliths; (2) sorption (distribution coefficient) and solubility parameter. This may range from a re-analysis of the DCF literature values, geochemical calculations to obtain material specific solubilities, to a limited number of experimental studies such as batch and leaching tests.

7. Conclusion

Long-term degradation of concrete EB in SA modelling was accounted for assuming a stepwise loss of the safety function ‘limitation of water ingress’. This was accomplished by developing models whose concrete components had increasing hydraulic conductivity values as degradation progressed. The models did not assume loss of the safety function ‘diffusion and retention’, owing to the large amount of cement present in the near field.

REFERENCES


Building confidence in the disposal safety case through scientific excellence: 
The OECD Nuclear Energy Agency Thermochemical Database\textsuperscript{1}

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Abstract. The OECD Nuclear Energy Agency (NEA) Thermochemical Database (TDB) is the product of an ongoing cooperative Project to assemble a comprehensive, internally consistent and quality-assured database of chemical elements selected for their relevance to the assessment of disposal safety. Major selection criteria for the inclusion of elements are mobility, radiotoxicity, inventory and half-life. The project is now in its 20\textsuperscript{th} year, and arose from the realization that existing databases lacked internal consistency or were not sufficiently documented to allow the tracing of the original data sources. This resulted in inconsistent results, e.g., from the same code, when using different databases for the same condition. Thus, increased confidence in the data was needed in order to take advantage, unequivocally, of the powerful insights provided by chemical thermodynamics in performing safety analyses. Confidence in the quality and applicability of the selected data is built upon, in the first place, through the adherence to procedures that are firmly established in the scientific community: formalized and traceable expert judgment, critical review by peers and open publication of both data and the process for their selection with possibility of feedback of experience and new insights. All procedures are specified in the Project guidelines that have remained essentially unchanged since the early stages of the Project. The effort so far has resulted in the publication of thermochemical data for eight elements comprising major actinides and fission and activation products. Added values of the Project are that (a) the thermochemical data are applicable to a variety of disposal systems and also for potential applications beyond disposal; (b) the forming of qualified personnel for the purpose of supporting safety assessment, which is an additional gage of confidence in the latter; (c) an efficient use of resources. For these reasons, the NEA TDB has evolved into a reference tool - utilised by all advanced national projects - for the provision of quality data for the safety assessment of disposal sites as well as a model for international co-operation in the field.

1. The need to increase confidence in thermochemical calculations

In order to assess the safety of a radioactive waste repository, it is essential to understand the geochemical behaviour of its components. Chemical processes such as the interaction of radionuclides with rock surfaces and coatings, or the formation of insoluble compounds, may greatly inhibit their migration into the environment. Numerical simulation and/or modelling of processes affecting the behaviour of radionuclides in natural and man-made systems is an integral part of a radiological assessment methodology. Some of the basic information is provided by speciation calculations using general, non-site-specific, chemical thermodynamic data. The value of the results of geochemical modelling as a predictive tool is strongly dependent on the quality of the thermodynamic data used to calculate the chemical speciation. The TDB project was initiated in 1984 as a joint activity of the NEA Data Bank and the Division of Radiation Protection and Waste Management. At the time, it was clear that although a number of thermodynamic data compilations and reviews had already been published [1-4], none of them could be used reliably as a complete source to study the behaviour of radionuclides in the environment. To be useful and to build confidence in the utilization of data in

\textsuperscript{1} The following organisations participate in the NEA TDB Project: ANSTO (Australia), OPG (Canada), NIRAS/ONDRAF (Belgium), RAWRA (Czech Republic), POSIVA (Finland), ANDRA (France), IPSN/IRSN (France), FZK INE (Germany), JNC (Japan), ENRESA (Spain), SKB (Sweden), SKI (Sweden), HSK (Switzerland), NAGRA (Switzerland), PSI (Switzerland), BNFL (UK), NIREX (UK), Department of Energy (USA).
performance assessment work, a database must: contain data for all the elements of interest in radioactive waste disposal systems; describe in a traceable documentation why and how the data were selected; document the sources of experimental data used; be internally consistent; and treat all solids and aqueous species of the elements of interest. None of the existing data bases at the time fulfilled all of these criteria. Critically, the documentation on how and why a particular piece of data was selected was (and in most cases, it still is) often omitted. It is also common to find specialised thermochemical databases intended for quite different purposes, such as general geochemical modelling under hydrothermal conditions, metallurgical simulations and others [5-8]. Consequently, most research groups supporting the performance assessment of radioactive waste disposal still use their own databases for modelling purposes. However, these individual databases may lack internal consistency, and they often differ considerably from each other, especially for actinide data. It is thus not surprising that radionuclide speciation and maximum solubilities calculated by different groups, with different geochemical computer codes and data, but for similar conditions, often turn out to differ by orders of magnitude. It has been recognised that these discrepancies are due to shortcomings in the different databases, rather than in the computer codes used [9].

This late 1980's scenario and its suite, described in subsequent sections of this paper, lends itself to analysis under the optics of developing and communicating confidence in the safety case, and more specifically in the geochemical calculations, as defined by the OECD NEA in the late 1990's [10]. Three additional circumstances should be mentioned here: first, an extensive corpus of experimental data exists in the literature that enables the selection of quality thermochemical data without needing to revert in many cases to new experiments; second, such selection requires a detailed, critical review of the literature sources and recalculations to bring all the reported properties to a common standard thermodynamic state so it goes well beyond the requirements of a “thorough compilation”; third, such work can only be performed by experts in many diverse physicochemical experimental and data analysis techniques. This latter type of expertise is starting to be less common in the academic world whose dynamics naturally bring it - in words of Lewis and Randall - to the “very borderland of the unknown” and not so much to transit the “broad highway of thermodynamics” [11].

2. TDB Phase I: Building confidence along three components

Following initial work [12-14], during the period 1984-1998 a comprehensive, internally consistent and internationally recognised thermodynamic database was developed at the NEA for the inorganic, aqueous and solid chemistry of five elements: uranium, americium, technetium, neptunium and plutonium (TDB Project Phase I). This database was backed by extensive reviews published in the open literature [15-18] and was applied to describe the behaviour of these elements under conditions relevant to radioactive waste disposal systems in the geochemical environment. The broad scope of the reviews and the fundamental nature of the data stored (Gibbs’ free energy, enthalpy, molar entropy and heat capacity for formation of species and reactions) allowed the utilisation of the databases for calculations applied to different waste disposal host formations (for instance, granitic or argillaceous environments for deep geological disposal concepts).

Currently we can identify already in the first published NEA TDB review [15], the Uranium report, the components along which NEA TDB helps build confidence in geochemical calculations to support safety assessments of studied disposal systems and which are widely proven in the scientific arena:

- Critical expert judgement of existing literature, review by scientific peers, detailed publication in the open literature;
- Knowledge transfer between TDB review teams and model implementers;
- Identification of areas needing further research.

Accordingly, the Uranium report, authored by a team of seven international experts and peer-reviewed by eight independent experts, not only contains extensive tables with values and uncertainties for thermodynamic properties for species and reactions involving uranium, but also detailed critical reviews of original experimental work explaining why the corresponding data were accepted (or rejected) for the final selection. A main chapter of the report is devoted to systematically presenting the recalculations leading to selected data and at the same time serving as an up-to-date and authoritative presentation of the chemical thermodynamics of each group of compounds of the element
under review. This latter type of presentation is particularly relevant as a guide to modellers since it allows them to implement geochemical calculations on the basis of authoritative, up-to-date chemical information for systems involving a large variety of aqueous complexes and solids limiting solubility. This constitutes an excellent vector for knowledge transfer between two specialist communities, since feedback can be established on the basis of the comparison of calculations resulting from the selected database and field experience. A bibliography and a list of authors serve as a catalogue of past and current expertise in the field. Supplementary material (essentially on data reduction to the thermodynamic standard state) is contained in Appendices, based in the Project guidelines. Finally, the database is made available on computer media through the NEA Data Bank. Thus, in terms of the classification established in the IPAG-3 exercise [19], a TDB review can be seen to contribute to the full set of arguments for building confidence in the data and knowledge of disposal systems.

3. Review procedure: The Project Guidelines

The procedure adopted for TDB reviews is specified in the Project Guidelines, essentially unchanged from the beginning of the Project and which detail its organizational and scientific aspects [20]:

- TDB-0: “The NEA Thermochemical Database Project”, contains an overall description of the Project organization and objectives;
- TDB-1: “Guidelines for the review procedure and data selection” deals with the steps to be followed by the review teams from the systematic review of all published sources of experimental thermodynamic data to the documentation of the final selections;
- TDB-2: “Guidelines on the extrapolation to zero ionic strength” is specially relevant for ensuring that data for aqueous species have been reduced to a common standard thermodynamic state;
- TDB-3: “Guidelines on the assignment of uncertainties”: assigning uncertainties to data is a major task for reviewers who have to (a) exert their expert judgement on the validity of uncertainty estimation by original authors or substitute for the lack of the latter by an ‘educated guess’ and (b) use statistical techniques to estimate the compatibility of data-sets originating in different laboratories and experimental methods;
- TDB-4: “Guidelines for temperature corrections”. Although a large amount of data exists in the literature, most of it refers to temperatures near 25 C. This Guideline deals with procedure to correct experimental measurement to 298.15 K and to estimate values of thermodynamic properties at, for example, higher temperatures as may be encountered in waste disposal systems;
- TDB-5: “Standards and conventions in TDB publications” deals with the definitions of the standard thermodynamic state, the symbols and terminology that have been adopted for the Project;
- TDB-6: “Guidelines for the independent peer review of TDB reports” established the procure for the review by independent experts of the TDB reports, once they have been produced by the TDB review teams.

One interesting aspect of these Guidelines is that they have been adopted in their essentials by several national and international organizations both within [21-24] and outside [25] the field of waste disposal for the critical evaluation of databases for other elements of their immediate interest.

4. Improving the paradigm: TDB Phase II and TDB III

Phase I of the TDB Project resulted in highly recognized critically reviewed data and set new international standards for the critical review of chemical thermodynamic data through international co-operation. Not only the high scientific standards but also the framework were appreciated by the radioactive waste disposal community: the more urgent objectives having been agreed based on the needs of the various national programmes, they were pursued by teams of independent scientists (from universities and research laboratories both in OECD member and non-member countries) under the direction of an International Organisation. However, important delays had been encountered.

A second phase of the TDB Project, was agreed upon by 17 organisations from OECD NEA member countries for the period 1998-2003. The new phase was organised as a semi-autonomous project under the guidance of a Management Board, representing the participating organisations. The OECD NEA
Data Bank acts as Project Co-ordinator, liaising with the Review Teams consisting of approximately three to five experts per team, approved by the Management Board. As in Phase I, the main tasks of the Review Teams are to critically and independently review the chemical thermodynamic data available for the element in question, recommend a set of data, and present these data, together with a justification for the selection, in a report. The members of the review teams are highly qualified in the area of science covered by the review. Their experimental experience is crucial for judging the quality and completeness of the scientific publications to be reviewed. The review areas were decided by taking into account the mobility, radioactivity and half-lives of the commonly occurring nuclides in radioactive waste, as well as the particular areas of interest of the participating organisations. In particular, the programme of work for Phase II gave priority to: updating the existing database for uranium, neptunium, plutonium, americium and technetium; developing new databases for other fission and activation products (selenium, nickel and zirconium) and for compounds of and complex of simple organic ligands with all the elements previously contemplated. It should be noted that extending the scope of the TDB reviews outside the actinides, makes them of interest to a broader community in environmental geochemistry.

To foster the exchange of information between those developing new thermodynamic data, either by direct experimental work in the laboratory or participation in the TDB review teams, and those modeling and analyzing field data, the TDB Phase II Project organized in 2001 a workshop, held under the auspices of the OECD NEA, on “The Use of Thermodynamic Databases in Performance Assessment”, which was attended by experts from both communities [26]. The workshop focused on two of the identified major problems: how to cope with the lack of experimental thermodynamic information for certain systems and the limited awareness in part of the scientific community about the precise database requirements needed in performance assessment. Two main conclusions of the workshop were the need to continue the TDB effort, in parallel with experimentation, to gain confidence by increasing ‘reality’ (i.e., describing systems such as minerals or solid solutions), increasing ‘reliability’ (for certain type of data such as redox properties of actinides) and increasing the ‘quantity’ of the data (under realistic safety case conditions) [21] as well as gaining confidence by training modelers in the use of the TDB reviews not only to extract data but also the chemical model information that they contain.

By 2003, the update of the database had been completed and other reviews had entered their final stage [27-31] and further needs were identified by the participating organizations. Thus, TDB III, a third phase of the Project was started in order to carry out the new programme of work under the same framework as TDB Phase II. Currently, in line with the recommendations from the workshop discussion, this third series of reviews is being prepared dealing with elements determining the chemical conditions in underground disposal systems (Fe), contributing significantly to the uncertainty in the estimation of the radiological impact (Sn) or needed to improve the physicochemical consistency of the database (Th). Furthermore, the Project provides a framework where guidelines for future review work are developed and presented to the radioactive waste disposal community, such as the description of solid solutions and their equilibria with aqueous solutions.

5. Conclusions and prospects for the future

Confidence in long-term disposal safety rests on the quality of the basis on which the assessment is made. This is turn rests on the quality of the models and data that constitute the assessment basis. Through its twenty years of existence, the NEA TDB Project has proved to be a successful international collaborative framework where the needs for critically reviewed chemical thermodynamics data for radioactive waste disposal have been met in a common international framework by the joint work of experts under strict criteria of scientific quality and traceability, subject to review by peers and leading to publications in the open literature. Thus, confidence in the quality and applicability of the selected data is built upon, in the first place, through the adherence to procedures that are firmly established in the scientific community: formalized and traceable expert judgement, critical review by peers and open publication of both data and the process for their selection with possibility of feedback of experience and new insights. The identified procedure and guidelines are robust, and have been adopted in their essentials also by several national and international organizations outside the field of waste disposal for the critical evaluation of databases.
for other elements of their immediate interest. In OECD member countries, these TDB reports are a reference point for the waste management disposal safety assessment, as well as a guide for future research directions. The NEA TDB database and its associated reports are available to the general interested public, which provides an additional vector for knowledge dissemination and preservation and for confidence building.

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The Mexican law on the safety of radioactive waste disposal

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Abstract. In this paper, we will briefly answer; using Mexican Positive Law, the following question: How does Mexico regulate radioactive waste disposal?, In order to do that we should focus both in Mexican entities or in Federal Public Administration Institutions involved in radioactive waste disposal, and in legal dispositions that deal with such activities.

1. Introduction
The legal system of the Mexican United States, is classified in the “Roman-Germanic” law family (Written Law), therefore, its legal ordinances are writings that come from the legislation process for law creation with the approval of the Union Congress, after a proposal of the Federal Executive Power, of the Deputies and/or Senators, or after a proposal of one of the State Congresses that are part of the Mexican Federation [1].

According to the hierarchy of the judicial norms pondered by the jurist Hans Kelsen in his work “Pure Theory of Law” [2], the maximum legal ordinance of Mexico is the Political Constitution of the Mexican United States [3], which states that:

- The generation of Nuclear Power is a strategic area exclusively reserved to the Mexican State, which does not constitute a monopoly [4];
- The Mexican State will have the necessary organisms to efficiently handle the strategic areas under its responsibility [5];
- The exploitation of nuclear fuels for Nuclear Power generation and their regulation for other purposes belongs to the Nation. The use of nuclear energy will be only for pacific purposes [6];
- The public sector will be exclusively responsible for the strategic areas pointed out in the article 28, fourth paragraph of the Mexican Constitution, while Federal Government keeps the property and control over the organisms established for the mentioned purpose [7].

2. Judicial hierarchy
Following the judicial hierarchy, under the Mexican Constitution, there are the International Treatments signed by Mexico, such as the Convention for the Prevention of Sea Pollution by the Disposal of Waste and Other Matters, issued by FOD [8] on 16 July 1975, valid since 30 August 1975.

From this convention it is worth citing its article 12 letter “d”; which states that the parties commit to adopt the necessary measures in order to protect the sea medium against pollution caused by radioactive pollutants no matter its origin.

Following the Mexican judicial hierarchy, we have the following Mexican laws:

- **Organic Law for Federal Public Administration** (FOD 29/XII/1976. Valid since 01/1/1977): The Secretary of Energy is entitled to: conduct the energy policy of the country; exercise the rights in nuclear power matters, conduct the activity of the para-state entities whose object is related to nuclear power exploitation and transformation, sticking to the legislation in ecological matters; regular and in its case, to issue official Mexican norms in nuclear safety and prevention, including that relative to its use, production, exploitation, profiting, transportation, alienation, import and export of radioactive materials, as well as to control and survey their due compliance [9].
• Regulation Law of the Article 27 from the Constitution in Nuclear Matters (FOD 04/II/1985, valid since 05/II/1985): Regulates exploration, exploitation, and the benefit of radioactive minerals, as well as the use of nuclear fuels, the use of the nuclear energy, research in nuclear science and techniques, nuclear industry and all that is related to it [10]. Nuclear industry is comprised by, among others, the last phases of the fuel cycle, including definitive and temporary storage of the irradiated fuel or of the radioactive waste derived from reprocessing, as well as the processing, conditioning and final disposal of radioactive waste [11]. The president of Mexico, through it is the Secretary of Energy’s responsibility the storage, transportation, and deposit of nuclear fuels and radioactive waste no matter of their origin [12]. The placement, design, construction, operation, modification, cease of operations, definitive shut down, and dismantling of nuclear and radioactive facilities requires the authorization of the Secretary of Energy [13]. The Nuclear Research National Institute is a public organism de-centralized from the Federal Government with judiciary personality and its own patrimony [14], which objective is to carry out research and development in the field of nuclear science and technology, as well as to promote the peaceful use of nuclear energy and to divulge its advances to link them to the economic, social, scientific and technological development of Mexico [15]. The Nuclear Safety and Prevention National Commission is a non-concentrated organ dependent of the Secretary of Energy; which is responsible for checking, evaluating and authorizing the bases for location, design, construction, operation, modification, cease of operations, definitive shut down and dismantling of nuclear and radioactive facilities; as well as for all relative to fabrication, use, handle, storage, reprocessing and transportation of nuclear materials and fuels, radioactive materials and equipment which contains them; processing, conditioning, pouring and storage of radioactive waste, and any other disposal of them [16].

Later, considering the judicial hierarchy, we have Mexican Regulations:

• General Regulation of Radiological Safety (FOD 22/XI/1988, valid since: 23/XI/1988). Its purpose is to provide in the administrative sphere the observance of the Regulatory Law of Constitutional Article 27 in Nuclear Matters in that relative to radiological safety. The topics included in this regulation are: system of dose limitation; application of the system of dose limitation; planned and emergency exposures; ionizing sources of radiation; open sources, radioactive facilities; conditions of the radioactive facilities; of the permissionary, responsible of radiological safety and exposed personnel due to its occupation; of the preventive measures or of security; of the authorizations, permits and licenses; of inspections, audits, verifications and acknowledgements and of sanctions.

• Interior Regulation of the Secretary of Energy (FOD 01/VI/1995, valid since: 02/VI/1995, the Secretary of Energy depends of the President of Mexico [17]. To deal with the issues concerned the Secretary of Energy has, among other administrative units, the Nuclear Safety and Prevention Commission [18]. The Secretary of Energy can authorize the location, design, construction, operation, modification, cease of operations, definitive shut down and dismantling of nuclear and radioactive facilities, as well as to command to the Nuclear Safety and Prevention National Commission the temporary occupation of nuclear and radioactive facilities that represent danger or risk to the workers or to the general population [19].

And finally, in the Mexican judicial hierarchy there are the Official Mexican Norms (NOM’s) over Radioactive Waste; which specify, with further details than other Mexican legal ordinance, the safe elimination of radioactive waste as described below:

• Classification of Radioactive Waste: (FOD 4/Marzo/1996). Its goal is to set the criteria for classifying radioactive waste that are produced by the nuclear industry. It states that radioactive wastes are classified according to the concentration, the activity and the mean life of radionuclides that are present in them and its origin as: radioactive waste of low level: Class A, Class B and Class C; Radioactive Waste of Intermediate Level; Radioactive Waste of High Level; Mixed Waste; Uranium and Thorium wastes.
Methods for determining the concentration of activity and total activity in the bulks of radioactive waste: (FOD 12/August/1996) to determine the concentration of the radionuclides and their activity in the bulks of radioactive waste in a reliable way before and after their treatment and/or conditioning, the methods listed below should be used, previous justification of their election: balance of materials, classification by source, total activity measurements and specific radionuclides measurement.

Requirements for bulks of radioactive waste of low level for shallow definitive storage: (FOD 14/August/1996). It is necessary that the constitutive elements of the bulks of radioactive waste show features relative to the radionuclides confinement; their strength under load; their resistance to thermal cycles; and their stability in the presence of radiations.

Requirements for radioactive waste incineration facilities: (FOD 15/August/1996). These facilities must comply with operational criteria and safety aspects; as well as design and construction requirements.

Lixiviation Tests for specimens of solidified radioactive waste: (FOD 4/August/1997).

Requirements for a facility for shallow definitive storage of radioactive waste of low level. Part 1: Site: (FOD 5/September/1997) points out the features that the site must have.

Requirements for a facility for the shallow definitive storage of radioactive waste of low level. Part 2: Design: (FOD 5/September/1997) points out the stability of the engineering barriers that must be considered in the following periods: for waste class A, 100 years; for waste class B, 300 years and for waste class C, 500 years.

Requirements for a facility for shallow definitive storage of radioactive waste of low level. Part 3: Operation and shut down: (FOD 14/January/1999) sets up the requirements for construction, the requirements for operation, the requirements for shut down, the requirements for post-shut down and the institutional control. In addition to a program of environment surveillance 1)pre-operational (not less than 24 months); 2)operational and post-shut down (this program should include the measurement and periodic re-evaluation for those physical parameters that show variations with time, such as wind speed and direction, atmospheric stability, precipitation, temperature and vaporizing) and 3) during the institutional control period (evaluation of the environmental impact in the long term).

Handling of radioactive waste in radioactive facilities that use open sources: (FOD 22/December/1998). Its goal is to set the requirements that must be observed during the administrative and operational activities involved in the handling of radioactive waste, in facilities that use open sources. Includes the handling that must be given to radioactive waste, depending on whether they are solid or liquid radioactive waste.

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[16] Article 50, fraction III.
[19] Article 5 fraction XX.

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   b) Methods for determining the concentration of activity and total activity in the bulks of radioactive waste.
   c) Requirements for bulks of radioactive waste of low level for shallow definitive storage.
   d) Requirements for radioactive waste incineration facilities.
   e) Lixiviation Tests for specimens of solidified radioactive waste.
   f) Requirements for a facility for shallow definitive storage of radioactive waste of low level. Part 1: Site.
   g) Requirements for a facility for the shallow definitive storage of radioactive waste of low level. Part 2: Design.
   h) Requirements for a facility for shallow definitive storage of radioactive waste of low level. Part 3: Operation and shut down.
   i) Handling of radioactive waste in radioactive facilities that use open sources.
Status and progress of development of radionuclide inventory declaration method for the disposal of radioactive waste in Korea

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Abstract. Regulations and guidelines require the detailed information about the characteristics of radioactive waste drums prior to transport to the disposal sites. In particular, the measurement of concentration and total activity of radionuclide contained in radwaste drum is very important for the accurate determination of source term and a reliable management of radioactive waste. As a part of effort to meet the waste acceptance criteria, waste generator is in progress to develop the radwaste assay system and the radionuclide declaration methods for the low and intermediate level waste generated from the Korean nuclear power plants. To obtain the database specific to Korea, a large number of sampling and radiochemical analysis are also in progress. From the database, the effort to find the most reliable radionuclide inventory declaration methods is continued. Moreover, the periodic validation method will be set up and additional work will be conducted to increase the reliability and to satisfy the regulatory requirement.

1. Introduction

In Korea, 20 nuclear power plants (NPPs) composed of 16 Pressurized Water Reactors (PWRs) and 4 CANDU Reactors (PHWRs) are currently in operation. The construction and operation of low and intermediate level waste (LILW) disposal facility is urgent because the cumulative amounts of low and intermediate level radioactive waste in NPP reached about 60,000 drums (unit: 200 liter) and the saturation of storage capacity for the LILW will be reached in 2008. Thus, the selection of radwaste disposal site is an pressing national project at present time. Regulatory body require an information about the radioactive wastes prior to the transport to the disposal site. Thus, the Enforcement Decree of the Korean Atomic Energy Act requires the Minister of Science and Technology of Korea to establish regulation for the waste acceptance. It requires detailed information about the radioactive waste package and its contents, especially the activity of radionuclides and total activities.

For the measurement of the concentrations and activities of radionuclides in radwaste drum, a radionuclides assay system was installed at Korean Kori site in 1996. Scaling factor (SF) methods and segmented gamma scanning method had been played a dominant role in this system. However, the work was conducted in a limited scope such as a target radionuclides and the number of sampling. Moreover, other PWRs except NPPs of Kori site and PHWRs are not considered. The Korea Institute of Nuclear Safety (KINS), regulatory authority, is under developing the more detailed waste acceptance criteria involving specific target radionuclides to be declared, its declaration method, limits of radionuclide concentrations, and so on. In Line with this efforts of regulatory authority, Korea Hydro & Nuclear Power Co., Ltd. (KHNP) is in progress to establish an radwaste assay system using more reliable methodology for updating the performance of Korean nuclear waste management. The KHNP organizes the overall project with partial co-operation with Korea Power Engineering Company Inc. (KOPEC), Korea Atomic Energy Research Institute (KAERI) and Korea Advanced Institute of Science and Technology (KAIST). In this paper, it is briefly introduced the current work and plan, which is conducting by the utility, to meet the waste acceptance criteria about the radionuclide declaration.
2. Status of radionuclide activity determination method in Korea

At the end of 1993, Korea electric power research institute (KEPRI) organized the project to design and install the radioactive waste assay system with partial co-operation with KAERI and KAIST [1,2]. With careful considerations, Kori NPP was selected as a candidate site for the assay system. The Radwaste assay system was installed and started operation during the mid 1996. In this research, activity determination was conducted by radwaste assay system for a gamma-emitting radionuclides and SF method for difficult-to-measure (DTM) radionuclides. In the SF method, the concentration of DTM radionuclides is estimated indirectly by relating DTM radionuclides to other easy-to-measurable key radionuclides. SFs are generated by use of sample data that are gathered from the radiochemical analyses of waste samples collected from the different waste stream. A further detailed research has been progressed for more reliable assessment and continuous renewal of database. The project of new radwaste assay system was started in 2003 and will be finished at the end of 2005. Principal changes of new system compared to Kori system are assaying target DTM radionuclides, target NPPs, more detailed classification of the waste stream, and the number of sampling. In addition, sampling of the long-term stored radwaste drums is included. Comparison between Kori system and new system is summarized in Table I [3].

### Table I. Comparison of Kori system and new system

<table>
<thead>
<tr>
<th></th>
<th>Kori System</th>
<th>New System</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gamma assay method</td>
<td>Based on the segmented gamma scanning (SGS) method</td>
<td>Based on the tomographic gamma scanning (TGS) method</td>
</tr>
<tr>
<td>Target DTM radionuclide</td>
<td>$^3$H, $^{14}$C, $^{55}$Fe, $^{60}$Co, $^{63}$Ni, $^{90}$Sr, $^{94}$Nb, $^{99}$Tc, $^{137}$Cs and Gross alpha</td>
<td>$^3$H, $^{14}$C, $^{55}$Fe, $^{60}$Co, $^{59}$Ni, $^{63}$Ni, $^{90}$Sr, $^{94}$Nb, $^{99}$Tc, $^{137}$Cs, $^{129}$I, $^{238}$Pu, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu, $^{242}$Am, $^{244}$Cm and gross alpha</td>
</tr>
<tr>
<td>Target NPPs</td>
<td>4 PWRs (Kori site)</td>
<td>16 PWRs and 4 PHWRs</td>
</tr>
<tr>
<td>Waste types (representative sampling)</td>
<td>A) Spent filter (RCS letdown filter) B) Concentrates (Evaporator bottom) C) Spent resin (Primary mixed bed resin) D) DAW (Dry active waste)</td>
<td>A) Spent filter (RCS letdown filter) B) Concentrates (Evaporator bottom) C) Spent resin (Primary mixed bed resin) D) DAW (Dry active waste) E) Sludge</td>
</tr>
<tr>
<td>Number of sample</td>
<td>24</td>
<td>320$^b$</td>
</tr>
</tbody>
</table>

$^a$ Reactor cooling system  
$^b$ Including the samplings from the RCS and the long-term stored radwaste drum

3. Development of activity determination method

The representative sampling is in progress and scheduled until the end of 2005. More than 60 % of scheduled sampling is currently conducted and the analysis of each target radionuclide is progressed for each sample. To meet the requirement of regulatory body, candidate radionuclide declaration methods, especially for the DTM nuclides, are studied from other countries’ experiences and are continuing to test each method by using the accumulated database. The interim conclusion is obtained from the case study for each nuclide. The candidate methods for the radionuclide inventory determination are summarized in Table II.
Table II. Candidate radionuclide declaration method

<table>
<thead>
<tr>
<th>Radionuclide declaration method</th>
<th>Target nuclide</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tomographic gamma scanning method</td>
<td>(^{60}\text{Co}, ^{137}\text{Cs}, \text{and gamma-emitting radionuclides})</td>
</tr>
<tr>
<td>Corrosion products(^a)</td>
<td>(^{3}\text{H}, ^{14}\text{C}, ^{55}\text{Fe}, ^{59}\text{Ni}, ^{63}\text{Ni}, ^{94}\text{Nb})</td>
</tr>
<tr>
<td>Scaling factor method</td>
<td></td>
</tr>
<tr>
<td>Fission products(^b)</td>
<td>(^{90}\text{Sr}, ^{99m}\text{Tc}, ^{129}\text{I})</td>
</tr>
<tr>
<td>TRU nuclides(^c)</td>
<td>(^{238}\text{Pu}, ^{239}\text{Pu}, ^{240}\text{Pu}, ^{241}\text{Pu}, ^{241}\text{Am}, ^{242}\text{Cm}, ^{244}\text{Cm}) and Gross alpha</td>
</tr>
<tr>
<td>Mean concentration method</td>
<td>(^{3}\text{H}, ^{99m}\text{Tc})</td>
</tr>
</tbody>
</table>

\(^a\) Key nuclide : Co-60

\(^b\) Key nuclide : Cs-137

\(^c\) Key nuclide : Cs-137 or Co-60

4. Discussion

As a part of work to meet the waste acceptance criteria, the joint project, which is organized by the utility, is in progress to establish an radwaste assay system using more reliable methodology compared with the existing radwaste assay system in Korea. To satisfy the regulatory body’s request, the target radionuclides are increased and the classification of waste stream is diversified for overall Korean 20 NPPs. In addition, the candidate radionuclide inventory declaration methods are established for each DTM radionuclides. Once the database is completed at the end of 2005, most reliable method will be determined and reflected to the radionuclide inventory determination program. However, additional efforts for the declaration of radionuclide inventory in radwaste packages are required, especially for the improving the accuracy. Moreover, additional periodic sampling should be conducted for updating the database and validation of the methods. As the study goes on, reliability of radionuclide declaration method based on the Korean analyzed data will be improved, and more accurate and reliable prediction of radionuclide inventory in radioactive waste will be possible.

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Site characterization methodologies for the safe disposal of high level radioactive waste in granite mass: A case study

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Abstract. The Beishan site, located in northwestern China’s Gansu province, has been selected as a potential site for China’s high level radioactive waste repository. Granite is the host rock in the area. Since 1999, systematic site characterization technologies have been used at the site, the performance and effectiveness of methods such as remote sensing technology, surface electromagnetic survey, diamond borehole drilling technology, borehole acoustic televiewer survey, borehole radar survey, hydro-geochemical logging and geo-stress measurement have been evaluated. Those methods proved to be effective and useful in the granite site.

1. Introduction

China had started the research and development programme for high level radioactive waste (HLW) disposal in 1985. The objectives of the programme are to establish capability for the final disposal of HLW and to build China’s HLW repository in about 2050.

Site selection, started in 1986, has been an important part in China’s HLW disposal program. The whole site selection process was divided into 4 stages: nation-wide screening, regional screening, area screening and site confirmation. During siting process, the following factors have been considered: social-economic factors and natural factors, including population, economical potential, plant/animal resources, mineral resources, land use, local public attitude, geological/hydro-geological conditions and engineering conditions.

Since 1990, most of the efforts have been concentrated to Beishan area, Gansu province, which is considered as the most potential site for China’s HLW repository. Studies include regional crust stability, tectonic evolution, lithological studies, hydro-geological studies, preliminary geophysical survey, borehole drilling and systematic borehole tests.

In the period of 2000-2004, systematic site characterization studies have been conducted in Beishan area. Four boreholes were drilled, site characterization methods were studied. The performance and effectiveness of methods such as remote sensing technology, surface electromagnetic survey, diamond borehole drilling technology, borehole acoustic televiewer survey, borehole radar survey, hydro-geochemical logging and geostress measurement have been evaluated, those methods have been proved effective and useful in the granite site.

2. Geology of Beishan site

The Beishan area is located in northwestern China’s Gansu province. It is a remote arid Gobi desert area, with very few inhabitants. The precipitation in the area is about 70 mm/a, while evaporation reaches about 3000 mm/a. Tectonically; Beishan is located in the eastern part of the Tianshan-Beishan folded belt in western China. The candidate host rock is granite. The crust in the area has a block structure, with a crust thickness of 47–50 km. The depth contour of the crust strikes nearly EW, with very little variation. The gravity anomaly is approximately \(-150 \times 10^{-5} \text{ to } -225 \times 10^{-5} \text{ m/s}^2\). The gravity gradient is less than 0.6 mGal/km. On the gravity-anomaly map, the gravity-anomaly contour is distributed very sparsely, without obvious step zones, indicating that there are no large faults extending to the depth of the crust. The seismic intensity of the region is less than 6, and no earthquakes with Ms>4 ¾ have occurred. The topography of the area is characterized by a flat Gobi and small hills, with elevations ranging between 1000 m and 2000 m. Variations in height are usually
several tens of meters. The region is slowly uplifting without obvious differential movement. Geological characteristics of the Beishan region shows that the crust in the area is stable, and it has great potential for the construction of a HLW repository.

In Beishan region, eight granite blocks have been selected as potential sites for the future HLW repository. Among them, three blocks (Jiujing, Xiangyangshan-Xinchang, and Yemaquan) have been chosen as the sites with the most potential, and detailed work is now concentrated on them.

3. Remote sensing technology

The Beishan area is a Gobi desert area, with very good outcrops of bedrocks and very little vegetation. These features provide good basis for the use remote sensing technology in surface geological mapping and hydrogeological investigation.

Satellite images such as TM (Thematic Mapper) images and SPOT images were used in geological investigation of the Beishan area. The composed image of TM band 7, 4 and 2 have been successfully used to identify the distribution of different rock types, dykes, faults, lineament and Quaternary basins in Jiujing and Yemaquan area. SPOT images have high geometric resolution, they are mainly used together with TM images for the interpretation of lineament and faults. With the help of TM and SPOT images, draft geological map can be made before field investigation, this greatly enhance the working proficiency. With the combination of satellite and DEM (digital elevation map), 3D surface site map can be produced.

FIG. 1: Electromagnetic profile of Shiyuejing fault in Beishan granite site.
4. Surface geophysical survey

Detailed surface geophysical survey was conducted in Beishan area in order to identify the bearing of faults and their depth. Electromagnetic survey was proven the most effective surface geophysical method in identifying faults in granite site.

A STRATEGEM resistivity profile measurement system was used in Beishan site, the results show that the fault zone in granite is characterized by low resistivity (<100 Ω·m). Figure 1 shows the resistivity profile of the Shiyuejing Fault, indicating the dipping angle (85°), depth (larger than 2 km) and width (60 meters) of the fault, which is quite consistent with the trenching and drilling observations.

5. Borehole drilling technology

The purpose of drilling for site investigation is to get necessary information on the suitability of a site, through progressive compilation and evaluation of data from drilling and subsequent measurement. The drilling and subsequent measurement entail some disturbance. Drilling water, lubrication oils and dirt may be introduced into the rock by drilling. So, it is important to choose proper drilling technology.

The BS01 and BS02 are the first 2 in the Beishan site. BS01 is a vertical hole with depth of 700 meters, it is drilled to evaluate the potential granite unit, while BS 02 an inclined one with depth of 500 meters, focused to evaluate the characteristics of the key NE-striking fault: Shiyuejing Fault. Diamond drilling, core-sample taking drilling, with pure water as drilling fluid, and wire sampling technology for core recovery were used for the 2 bore hole drilling, which have been proved successful. For BS02, pipe casing, mud drilling method were used in order to go through the NE-striking Shiyuejing fault zone. By using this method, the borehole was successfully completed, perfect core samples for the fault have been obtained, with core recovery over 95%.

6. Borehole hydro-geological test - injection test

Granite is a fracture media, one of the most important features is the permeability of the rock mass and the fractures. Injection tests can be conducted in selected intervals in completed boreholes in order to obtain the parameters of permeability. Usually there are 2 types of injection tests: transient injection test for intact rock interval and normal injection tests for fractures. The length of measurement intervals is about 8 meter. In BS01, 6 intervals with fractures were chosen to carry out normal injection tests by using double packer system, while 4 intervals of intact rock were selected for transient injection test. The results of normal injection tests show that the permeability for the fractured zone ranges between 1.74×10^{-6} and <8.0×10^{-7}, while for intact rock between 1.85×10^{-9} and 2.66×10^{-8}, this shows the low permeability of the granite media. The experiences in Beishan site shows that injection tests could be the best tests to obtain the permeability of deep granite formation.

7. Borehole acoustic televiewer survey

A high resolution acoustic borehole televiewer system (Fac-40, made by DMT-ILG company of Germany) was used in BS01, in order to investigate the features of fractures and the integrity of the rock mass along the borehole. The accuracy of the system is 1%, while the resolution for fractures can be down to 0.1 mm, which is very powerful for the study of fractures. In BS01, the interval between 60 and 550 meter deep was investigated by the system. With the images obtained from borehole televiewer, the directional information of borehole wall, borehole deviation, fracture bearing, core orientation and other important data be obtained. All the data obtained during measurement can be processed by WellCAD software, and the statistics of fractures and the distribution of fractures can be obtained. For example, in an image for BS01, a fracture with its azimuth of 192.5° and tilt of 69° can be clearly seen at the depth of 425.9 meters. The statistic of fractures, resulting from the Fac-40 televiewer survey, reveals that the fractures can be divided into 2 groups: NE-striking and NW-striking. Most of the NE-striking fractures occur below the depth of 230 meter, but with 2 opposite
azimuth: NW and SE. Above 230 meters, most of the fractures are NW-striking. The average fracture density is about 8--12 per 10 meters. In a granite site, acoustic borehole televiwer measurement should be an essential way to study the fractures in borehole.

8. **Borehole radar survey**

A RAMAC bore hole radar survey system, produced by the Mala GeoScience Company in Sweden, was used in borehole BS01 in Beishan site. It is the first time to use such equipment in China. BS01 is the first borehole for China's site characterization program. During measurement, the dipole reflection mode was conducted by using the following parameters: Single reflection mode, 100 MHz antenna, antenna centre separation: 2.9m; stacks: 32; time window: 1043 ns; sampling frequency: 1007; distance interval: 0.2m. Results show that the penetration depth of radar wave in the borehole is about 20 meters and there are 22 reflectors cutting the borehole. The results are well consistent with the data obtained by geological logging, bore hole television survey and other geophysical survey. The practice has proved that the bore hole radar system is an effective tool to understand the extension of fractures and the integrity of rock mass, which is essential to site characterization program for high level radioactive waste disposal.

9. **Hydrogeochemical logging**

Deep geological environment in granite is vital to evaluate the suitability of the site. By using hydrogeochemical logging along boreholes, the deep environment parameters such as temperature, pressure, redox potential, pH, dissolved oxygen, conductivity and salinity can be obtained. During the measurement in Beishan, a Mont Sopris logging system with probes of temperature, pressure, redox potential, pH, dissolved oxygen, conductivity and salinity were used, and the results have shown such methods are useful to obtain those parameter. In BS01, measurement was conducted in the section of 0 to 500m, while for BS02, 0 to 300m deep, BS03, 70-490 m deep. The results have reveal that the deep environment is characterized by reducing, high salinity, neutral features. By analyzing the measurement curves, the water bearing zones and their outflow of water can also be identified.

10. **Geostress measurement**

Hydro-fracturing method was used to measure the in situ stress at borehole BS01 and BS03 in Beishan site. The major results shows that the maximum lateral principal stress ranges between 7.72 MPa in the shallow and 25.66 MPa in the deep, and has a direction range between N25°E--N45°E. The practices in Beishan shows that hydro-fracturing method could be the most suitable one to measure geostress.

11. **Conclusions**

Site characterization practices in Beishan site, Gansu Province, northwestern China, have proved several useful and effective methods for granite site such as: remote sensing technology, surface electromagnetic survey, diamond borehole drilling technology, borehole acoustic televiwer survey, borehole radar survey, hydro-geochemical logging and geostress measurement. By using those effective methods, some key parameters for site evaluation, concept design and performance assessment can be obtained. However, it is still necessary to develop further cost-effective and high-tech methods for site evaluation, in order to ensure the accuracy and the effectiveness to obtain site data.
Waste management policy and strategies for all waste types and disposal options and international co-operation - Bangladesh perspective

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Abstract. In Bangladesh, LILW radioactive wastes are generated from operation and maintenance of nuclear installations: Research Reactor, Radioisotope Production Laboratory, Neutron Generator, hot facilities, application/use of radiation sources (RS) in medicine, research, agriculture, industry. The radioactive wastes arising are described in this paper. The national policy concerning radioactive wastes and the activities being performed by the Health Physics and Radioactive Waste Management Unit of AERE, Savar, are outlined. BAEC has a perspective plan (2001-2010) to establish a pilot-scale near-surface waste repository for short-lived LILW. The factors being studied are presented, covering: general site description within the AERE campus; geology, hydrology and rivers; seismicity; surface drainage and ground water table; climate and meteorology; transportation and communication; demography and nearby facilities; vegetation; water and power supply; background radiation, nearest township; natural environment, land and water usage; dispersion of radioactive materials through groundwater; engineered feature.

1. Introduction

Solid, liquid and gaseous wastes (LILW) are generated from:

- operation and maintenance of different nuclear installations e.g., 3MW TRIGA Mark-II Research Reactor (RR), Radio-isotope Production Labs. (RIPL), Neutron Generator (by D-T reaction), the allied hot facilities, and the likes;
- application of radiation sources (RS)/radioisotopes (RIs) in medicine, research, agriculture, in industry.

The wastes arising are:

- spent ion-exchange resins, graphite plug, Pb-plug, PE-plug, RTD, solid trashes, rabbit system, etc. from RR;
- solids (e.g., contaminated glass vials, plastic syringes, hand gloves, tissue papers, protective cloths, etc.) and aqueous liquids from medical uses; spent sealed radiation sources (SRS) from industrial, medical (therapy) & research uses of RS, neutron generator;
- irradiated Al-vials, shoe-covers, contaminated articles, glass & plastic wares, activated carbon and liquids from production/processing and QC of Mo-99, Tc-99m, I-131;
- Gaseous wastes (Ar-41 from RR, I-131 from RIPL passes through activated carbon);
- solid and liquid wastes from labs. of research centers.

The radionuclides involved are: e.g., Co-60, Cr-51, Mn-54, Cs-134, H-3, Tc-99m, I-131, 125, Sr-90, C-14, Cs-137, Ir-192, Ra-226 (used/spent), Am-Be neutron source, etc.

2. Waste management policy

The national policy is to provide adequate protection for man and the environment against undue exposure to ionizing radiation from radiation sources (RS) and radioactive wastes (RW) for the present and future generation. The Nuclear Safety and Radiation Control Act '93 (Act. 21 of 1993) was duly approved and enacted by the parliament (1993) and the Regulations have been put into force on the 18\textsuperscript{th} Nov. 1997. The Health Physics and Radioactive Waste Management Unit, AERE, Savar has been performing the waste management (WM). The Nuclear Safety and Radiation Control Division is responsible regulatory control related with safe management of radioactive waste.
3. Waste management strategy (collection, temporary storage, processing, interim-storage, and disposal)

A mixed strategic system of: on-site collection, temporary storage (delay-and-decay) of short-lived wastes (e.g., NMCs), on-site collection & packaging of spent sealed radiation sources (SRS) & transportation to the centralized facility (CWPSF), and one-site collection, segregation, & transportation of RWs from the nuclear facilities of AERE to the CWPSF is being followed. R & D and relevant technological efforts on processing and disposal in progress are as follows:

(i) Standard and real liquid wastes containing Cs and Co radionuclides have been studied for their decontamination by chemical pptn. and ion-exchange-cum-Ultra filtration,

(ii) Ion–exchange-cum-ultra filtration plant treatment study (for Cs-137 & Co-60) has been performed with satisfactory test results,

(iii) Conditioning of Cs (by cementation) has been initiated,

(iv) About 1 curie of spent Ra-226 sources have been collected, characterized, encapsulated in ss capsules, placed in lead shielding devices, placed in the pre-fabricated IAEA standard 200L capacity MS drums and safely stored in the AERE campus, Savar (2000) under the IAEA Model Project "Sustainable technologies for managing radioactive waste”.

(v) Construction of the proposed CWPSF within AERE has been implemented. All non-conditioned & conditioned wastes would be collected, transported, processed as needed, and safely stored within the CWPSF.

4. Description of the site investigation for a repository

BAEC has a perspective plan (2001-2010) to establish a pilot-scale near-surface waste repository for short-lived LILW within the AERE campus. The technical and other factors being studied cover the following:

(a) General site description, existing facilities/installations within AERE campus:

The site (AERE) covers a total area of 256.0 acres (1.0 Sq. km) of land and stands on the Savar-Kaliakoir link road at a distance of about 40 km north-west of the capital city of Dhaka. The elevation of the site above sea level is about 10.7 m. There is a good road connection of the site with the city and other important places.

(b) Geology: The AERE is located on the highland part of the Madhupur Terrace. A 12-13 m silty-clay formation (Madhupur Clay) unconformably overlies a more than 90 m thick sandstone formation, which is the main aquifer of this region.

(c) Hydrology and rivers: The study area is bounded by three rivers - the Bansi to the west, the Turag to the east and the Dhaleshwari to the south. The flow direction of ground water in this area is from North-West to South-East. The average annual surface water level fluctuation in this region varies from 3.6 to 5.5 metre. Ground elevation ranges from 9.2 to 15.5 metre AMSL. Water table lies at depths of 0.30 to 12.20 m from the ground level depending on location, season of the year.

(d) Seismicity: Bangladesh is surrounded by the regions of the high seismicity. Experienced several great historical earthquakes during last 250 years and have small earthquakes occasionally (the following table presents a list of seven great earthquakes having magnitude >7).

Table 1. List of great historical earthquakes in and around Bangladesh

<table>
<thead>
<tr>
<th>Date</th>
<th>Name</th>
<th>Epicentre</th>
<th>Magnitude (M)</th>
</tr>
</thead>
<tbody>
<tr>
<td>10th Jan., 1869</td>
<td>Cachar Earthquake</td>
<td>Jainta Hill, Assam</td>
<td>8.9</td>
</tr>
<tr>
<td>14th July, 1885</td>
<td>Bengal Earthquake</td>
<td>Manikganj, Bangladesh</td>
<td>7.0</td>
</tr>
<tr>
<td>12th July, 1897</td>
<td>West Assam Earthquake</td>
<td>Shillong Plateau</td>
<td>8.7</td>
</tr>
<tr>
<td>18th July, 1918</td>
<td>Srimangal Earthquake</td>
<td>Srimangal, Sylhet</td>
<td>7.6</td>
</tr>
<tr>
<td>3rd July, 1930</td>
<td>Dhubri Earthquake</td>
<td>Dhubri, Assam</td>
<td>7.1</td>
</tr>
<tr>
<td>15th Jan., 1934</td>
<td>Bihar Earthquake</td>
<td>Bihar, India</td>
<td>8.3</td>
</tr>
<tr>
<td>15th Aug., 1950</td>
<td>Assam Earthquake</td>
<td>Assam, India</td>
<td>8.4</td>
</tr>
</tbody>
</table>
(e) **Surface drainage and ground water table:** In the western half, it drains the water through the existing channels running to the Bansi river (5.0 km) and in the eastern half water drainage find its way in the same manner to the river Turag (12.7 km). A number of bore-holes have been drilled in the campus (AERE) for observing groundwater level round the year. The range of water table is from 0.30m to a 7.10m below the ground level.

![Groundwater level fluctuations in the AERE campus (period 1990-97).](image)

**FIG. 1:** Groundwater level fluctuations in the AERE campus (period 1990-97).

(f) **Climate and meteorology:** AERE campus and its adjoining areas have a tropical humid climate with high summer temperature, excessive air humidity, heavy rainfall, and cool dry winter. The mean temperature of July is 27.78°C and that of January is 28.33°C. The average annual rainfall is about 2122 mm. The area is occasionally affected by the tropical cyclones and nor-westers associated with tornado.

(g) **Transportation and communication:** The site has a good road communication with the city and its surroundings. The site is located on the Savar - Kaliakoir link road 25 miles north-west of the city. The Zia International Airport is at 30 miles from the site. Industrial places at Tongi and Joydevpur are also connected to the site by the Savar-Kaliakoir link road which joins the Dhaka- Tangail road near Kaliakoir at about 7 miles (11.2 kms) north. The nearest railway station is at Tongi about 25 miles (40 kms). The river Bansi flows by the Savar Bazar which is located about 8 miles (12.8 kms) from the site.

(h) **Demography and nearby industrial, military and other facilities:** The population densities show quite a variation. The approximate population around the site within different radii up to 5 miles (8 kms), as per 1991Census, are shown below (Table-2).

<table>
<thead>
<tr>
<th>Radial distance from the Centre of the site</th>
<th>Number of Population Centre (approximate)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1/8 mile (0.2 Km)</td>
<td>600</td>
</tr>
<tr>
<td>1/3 mile (0.5 km)</td>
<td>2,000</td>
</tr>
<tr>
<td>1 mile (1.6 Kms)</td>
<td>250,000</td>
</tr>
<tr>
<td>5 miles (8 Kms)</td>
<td>400,000</td>
</tr>
</tbody>
</table>

In 1994, the total population within the radius surrounding the AERE was 1508,300. In the AERE housing coloney, it is about 1000.

(i) **Vegetation:** The site is not a very vegetative area. The vegetations commonly found in the adjoining areas are of ever-green type. The government plantation, the Bhabal Forest, is at 20 miles (32 kms) to the north and north-east and Modhupur Forest at 30 miles (48 kms) to the north-west.
(j) Water and power supply: The separate establishments have their own water supply systems. A Solar Pump (1.5 KW) has been installed at the AERE discharging with a capacity of 20,000 gallons of water per 8 hrs a day. The electric power supply to the site is being made from the 11 KV transmission line of Rural Electrification Board (REB). There are 1000 KVA and 500 KVA electric sub-stations to supply power throughout the site. For emergency, an additional stand-by Diesel Generator has been installed.

(k) Background radiation survey: Pre-operational and operational background radiation surveys have been performed. The results obtained are comparable and satisfactory.

(l) Nearest township: The nearest township is at Savar located at 8 miles.

(m) Natural environment, land and water usage: The land surface is small elevated plateau or table lands, is mostly used for agricultural and related purposes, various industrial units. The rest land is devoted to commercial, residential and recreational purposes. Ground water is drinking, irrigation and industrial. The river water is used for navigation, irrigation and fishing.

(n) Dispersion of radioactive materials through groundwater and surface water: Both ground water and surface water hydrology have been studied in order to address the problem related to the dispersion of radioactive materials. Various properties such as hydraulic conductivity (0.40 - 1.40 cm/day), porosity (37-41%), hydraulic gradient, ground water velocity (1.05-1.3) x10^9 cm/sec have been determined. The dose on humans due to the release of radionuclides from the experimental-cum-demonstration shallow land disposal facility was well below 0.25 µSv per year.

(o) Engineered Feature: The type of LILW repository will depend on

- the geological, hydro-geological and climatic conditions of the country, and
- specific disposal requirements and regulatory approaches.

5. Conclusion

Based on the preliminary site investigation, a multi-barrier engineered near-surface waste repository is being considered, and efforts are thus given for data generation. International technical co-operation is required in order to construct a multi-barrier engineered near-surface waste repository for disposal of radioactive waste.
Regulatory safety concerns on radioactive waste management in Chile

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Abstract. Chile has not specific radioactive waste management regulations, nevertheless some amount of radioactive waste are generated and treated in the country. The work and efforts by the regulatory authority, to complete and up-date the legal framework, according to the possible scenarios of future radioactive waste generation, is described.

1. Introduction
The generation of radioactive waste in Chile mainly arise from medical and industrial applications and some research facilities, like a research reactor (Chile has two research reactors, one in operation once a week, and the second one in extended shutdown, 5 and 2 MWt respectively), and a small Material Testing Reactor (MTR) fuel fabrication plant. Up to now, the spent fuel from the research reactor was sent to USA, according to an international agreement, but in the future the spent fuel discharged from the reactor will have to be treated in Chile. It is also foreseen that in the near future (year 2010), nuclear power plants will be constructed in Chile, with the consequently waste generation.

Nowadays, Chile has a waste conditioning plant capable to handle almost all medical and industrial radioactive waste generated in the country, and a temporary storage facility for conditioned waste, but this situation will change dramatically, when the need for treating the new MTR fuel elements fabricated in Chile will arise.

2. Regulatory actions
The Chilean authority for regulating radioactive waste management facilities is the Chilean Nuclear Energy Commission (CCHEN). The Department of Nuclear and Radiological Safety is responsible for proposing legal bodies to CCHEN Council, and the Safety of Radioactive Facilities Section is responsible for the safety evaluation and control of such facilities.

2.1. Legal framework
The efforts of the regulatory authority to have a working regulatory framework include the following actions:
(a) to establish a national policy on radioactive waste management, including waste coming from the nuclear cycle (spent fuel);
(b) to establish dispositions in an updated Nuclear Safety Law, regulating in a more precise manner, the management of radioactive waste in the country;
(c) to establish regulations for management of radioactive waste;
(d) to establish specific standards for management of radioactive waste.

2.1.1. National policy
The issue of a national policy establishing long term strategies needs a discussion to a national level, including political issues, and it is not foreseen when there will be a draft of this policy. This matter is also one of the main issues when discussing the option to build nuclear power plants in Chile, then it is a “must” to have the radioactive waste management policy online as soon as possible. Efforts will be dedicated to promote discussion at political levels and drafting a national radioactive waste policy.
2.1.2. Nuclear Safety Law

The updating of the Nuclear Safety Law is an ongoing work which was required to correct some international and legal problems introduced by a recent (2002) modification to the law. This opportunity will be used to reflect current country situation, and to include IAEA recommendations and security considerations applied to radioactive materials. Upgrading of Regulatory Body independence will be proposed in this revision. Also, the establishment of waste management criteria and provisions from the Joint Convention will also be included in this updating. Target milestone for sending the new law to the national congress is the end of 2005, and entry into force is expected by the end of 2006.

2.1.3. Regulations

The drafting of radioactive waste regulations is one of the main concerns of the competent regulatory authority, and the main efforts have been oriented to have the regulations ready along with the Nuclear Safety Law amendment. The basis for drafting the regulations has been the recommendations of the IAEA.

The scope of the regulations will be limited to the present classes of waste generated in Chile, and it is a clear need for including open provisions for taking into account the new types of waste that is expected to arise.

Some areas of regulatory concern are:
- clearance levels for radioactive waste;
- disposal or reuse of radioactive materials;
- mixed wastes, i.e. radioactive and other hazardous characteristics;
- uranium from the mining of non radioactive materials;
- storage of spent fuel at research reactors and the long term storage of spent fuel;
- ultimate solutions for management of spent fuel and radioactive wastes, taking into account the low amount generated in the country
- security and environmental protection
- resources needed to provide effective regulation
- difficulties in assuring an adequate availability of qualified staff
- funds to finance decommissioning and the consequential management of radioactive waste
- legacy situations resulting from previous practices
- regional solutions?

2.1.4. Standards

It is also foreseen that at least one standard on radioactive waste management will be issued the next year, establishing specific rules for treating the present type of waste. The effort to draft this standard will be initiated during year 2006 once there is a final draft of the regulations (2006).

2.2. Regulatory efforts

Besides the resources devoted to issue the legal bodies regulating the waste management, the regulatory body has two main challenges:

(a) to define effective licensing and control procedures for management of radioactive waste facilities, appropriate to all the possible facilities that is expected to be build in the country according with the national policy; and

(b) to prepare its personnel for carrying out the evaluation and control of the facilities associated to the radioactive waste management.

The assessment of present radioactive waste management do not pose a real challenge to the capabilities of the regulatory body, but assessing final waste disposal activities will impose a heavy regulatory load, beginning with the decision about the assessment approaches, safety evaluation of design characteristic of the final storage, taking into account its life span.
The assessment of spent fuel management is another subject of real concern and the actual regulatory personnel has no experience on this subject.

2.3. International scenario

The coming into force of the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management, that Chile has not yet signed and surely will signed in the near future, will place additional obligations to the regulatory effort that have to be taking into account in outlining present obligations and future developments in the regulatory field.

REFERENCES


Long-term storage and disposal of radioactive waste from the Chernobyl nuclear power plant

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Abstract. This paper describes the radioactive waste management activities at the Chernobyl Nuclear Power Plant (NPP). It shows that at present, the Chernobyl NPP site has accumulated a considerable amount of liquid and solid radioactive waste which is to be treated and handed over for disposal, and long-lived waste shall be handed over for temporary storage. Waste storage and disposal facilities design are described. The main RAW sources have been marked for which it is necessary to establish appropriate disposal and long-term storage sites.

1. Radioactive waste treatment facilities

At present, only short-lived low- and intermediate level solid radioactive waste (LILW-SL) is directly handed over for disposal. Annually, 1000-3000 m³ of such wastes are handed over for disposal. It is significant to note that according to the Ukrainian standards, only an enterprise that does not generate radioactive waste has the right to dispose of radioactive waste. There is the only disposal facility currently in operation at the Exclusion Zone, where the trench method of radioactive waste disposal is used.

At present, Liquid Radioactive Treatment Plant, LRTP, and Industrial Complex for Solid Radioactive Waste Management, ICSRM are under construction. Commissioning of LRTP in 2006 will allow to gradually transform liquid radioactive waste into solid form by cementing after preliminary evaporation of vat residue and centrifugation of pearlite and ion-exchange resins. The following operations shall be performed on ICSRM (approximate date of commissioning is 2007):

- recovery of Solid Radioactive Waste from the existing surface waste facility and transportation to the plant;
- acceptance at the plant of solid radioactive waste from different sources and their sorting in high-level waste, long-lived low and intermediate level waste, short-lived low and intermediate level waste. The first two classes of waste will not be treated, but shall be fragmented, in case of need, and then they shall be packed into 165 L and 200 L drums. The drums with high-level, and as well with long-lived low and intermediate level waste will be transported to the surface temporary storage facility (building is located at the Chernobyl NPP site) in reinforced-concrete containers, up to 8 drums a day.
- sorting of short-lived low and intermediate level waste in combustible, compressible and non-compressible waste;
- incineration of solid RW, adding combustible liquid RW;
- packing of compressible RW and incineration waste into 165 L drums;
- compaction of drums;
- placing of drum “pucks” into non-reusable reinforced-concrete 3.0 m³ containers and their cement grouting;
- placing of non-compressible RW directly into non-reusable reinforced-concrete 3.0 m³ containers and their cement grouting;
- transportation of non-reusable reinforced-concrete containers to the near-surface temporary storage facility, up to 3 containers a day.
2. Long-term storage and disposal of radioactive waste

Chernobyl NPP radioactive waste is managed in conditions of:
- after the accident at the Unit 4 of Chernobyl NPP in 1986 Chernobyl NPP site and immediate
  surroundings were significantly contaminated by radionuclides;
- in the Ukraine only solid radioactive waste is permitted to be disposed;
- in the Ukraine long-lived radioactive waste must be disposed in stable geological disposal
  facilities;
- the Ukraine has no burial (disposal) ground to dispose RW in stable geological facilities and the its
  establishment by estimation of the National Academy of Science of Ukraine will require as a
  whole no less than 600 million USD and no less than 10-15 years for development and survey
  works.

A near-surface disposal facility for LRTP and ICSRM products is built at a distance of 20 km from the
Chernobyl NPP and designed for disposal of 5632 containers of 3.0 m³ and 70 070 drums of 200 L,
that will allow to receive LRTP products during the period of not less than 8 years and ICSRM
products during the period of not less than 11 years. The disposal facility being built is designed to
receive radioactive waste within 30 years and then to store radioactive waste within 300 years, taking
into account regulatory inspection.

The RW disposal facility (Figure 1) is a surface construction in 273.1 × 44.1 m, consisting of two
sections located in parallel (plan size 273.1 × 18.8 m) and central drainage gallery, located between
sections underground. The sections contain 22 compartments made of monolithic reinforced-concrete
to dispose RW packages (2 sections, each has 11 compartments). The disposal volume is 71 280 m³.
Solid RW acceptance criteria for near-surface disposal facility provide receiving for disposal of
radioactive waste of LILW-SL category.

The existing re-equipped cells for storage of SRW of liquid solid waste storage facility (LSWSF) at
the Chernobyl NPP site will be used as temporary storage for HLW and LILW-LL which can’t be
disposed in the near-surface storage facility. HLW and LILW-LL will be sorted out, placed and stored
in 165 L drums which, in their turn, are placed into the outer packages (200 L drums with leak-proof
lids).

LSWS facility is a construction on the ChNPP industrial site (Figure 2) which is made of precast-
monolithic reinforced concrete. The facility has a rectangular configuration on the plan, dimensions in
axes are 66 x 69 m. Facility height is 30 m.

LSWSF provides for all required equipment for lifting and unloading of shipping containers and
placing of packages with HLW and LILW-LL for temporary storage in the reinforced concrete
shielded cells. LSWSF design capacity for LILW-LL and HLW is:
- no less than 6360 drums with HLW;
- 6784 drums with LILW-LL.

Maximum specific activity of RW accepted for storage by LSWSF is determined taking into account
protective properties of walls and cells coverings for SRW storage.
FIG. 1: Facility for near-surface disposal of Radwaste (LILW-SL) from ChNPP
3. Conclusion

The place of near surface temporary storage of HLW and LILW-LW is planned to accommodate about 13 000 primary packages, that will allow to receive ICSRM products within 10 years. Then, after a temporary storing of radioactive wastes of the above mentioned categories during 30 years, a decision will have to be taken to establish a final geological disposal facility or, after the second characterization, the waste can be transported to another intermediate radioactive waste storage facility.

It is necessary to point out the following important SRW sources for which it is required to establish appropriate disposal sites or long-term safe storage facilities:

1. Reactors graphite cladding, the mass of which for 3 units is more than 5000 t, is LILW-LL. The most acceptable decision now is a safe enclosure and following graphite cooling within the reactor cores for a period of not less than 100 years.

2. Power units 1, 2, 3 fuel channels are low-temperature HLW and will be low-temperature HLW during the long period. There are several variants to manage them:
   - FC safe enclosure and following cooling within the reactor core for a period of not less than 100 years;
   - removal of the FC from the reactor core after de-fuelling with following cooling in the reactor cooling ponds;
・ removal of the FC from the reactor core after de-fuelling with following compaction and handing-over for intermediate storage and disposal.

3. When establishing a final geological disposal facility for HLW and LILW-LW, it is necessary to take into account that after having been cooled in a dry spent nuclear fuel storage facility (ISF-2), additional intra-reactor absorbers and construction elements of fuel assemblies free from fissionable materials are subject to disposal.

4. Removal of RW and fuel containing materials (FCM) from the damaged Unit 4 is the most important problem, the solving of which supposes the construction of the New Safe Confinement (NSC), equipped with the proper technical equipment.
Classification of the inventory of spent sealed sources at Inshas storage facility and its regulation for disposal

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Abstract. Sealed radioactive sources have been widely used for many decades in industry, medicine and research. Although most countries have laid down a regulatory framework to control sealed sources, there are still a number of uncertainties concerning management of historical 226-Ra alpha sources and the possibility of retrieving non-registered sources. Both these uncertainties may represent high radiological risks for the population. In addition, management schemes and practices implemented in different countries can be somewhat conflicting and create problems for storage and disposal. This study has been performed to classify the existing sources at storage facility and then consider the situation relating to the regulation and management of sealed and spent sealed sources in Egypt, Czech Republic, Hungary, Estonia, Poland and Slovenia. Comparing between the existing Egyptian situation, IAEA Code of conduct on the safety and security of radioactive sources and the situation of these countries we can prepare develop regulatory recommendation package to the government of Egypt addressing all aspects of sealed sources management in Egypt. The joint project Integrated Management Program for Radioactive Sealed Sources (IMPRSS) between the government of Egypt and Sandia National Laboratories U.S., funded by the U.S, Agency for International Development and implemented through the U.S. Department of Energy have been performed and one of the objective of this project is developing the Egyptian regulations.

1. Introduction

The use of sealed radioactive sources world-wide has been increase in different activities such as in industry, medicine and research. The full life-cycle of these source from manufacture to disposal incorporate high radiological risks for the population. Therefore, most countries have established a regulatory framework to control the management of sealed sources. In Egypt, spent sources are stored in prepared rooms at storage area at Inshas site. According to IAEA classification that depending on the radiological properties of source during use, transportation, handling and storage of the sources, the stored spent sources are considered in all categories specified (1, 2, 3, 4, and 5) [1,2,3]. The aim of this study is to improve the Egyptian regulations relating to the management of sealed sources by comparing with Czech Republic, Hungary, Estonia, Poland and Slovenia regulations. Integrated Management Program for Radioactive Sealed Sources (IMPRSS) is a joint project between Sandia National Laboratories, U.S. and the Government of Egypt, funded by the U.S. Agency for International Development and implemented through the U.S. Department of Energy.

2. Spent sealed sources at Inshas site

There are about 595 spent sealed sources at Inshas site in the storage rooms [4]. The inventory of spent sealed sources contains 10 different sources Co-60, Cs-137, Ir-192, Sr-90, Am/Be, Ra-226, Am-241, Cd-109, Fe-55 and Kr-85. These 784 sources are classified according to IAEA categorization into the five categories.

2. The current Egyptian regulations

The Egyptian regulations are based on the Law No. 59/1960, the Law No. 4/1994 and the Decree Number 204 of 2000 and their implementations. These regulations covered some areas that related to the in-use SRs such as [5]: The Regulatory Bodies, Licensee responsibilities, Penalties, Dose constrains and Transporting of the radioactive sources.
Table 1: Classification of spent sealed sources at Inshas Site according to IAEA categorization

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Category</th>
<th>Unknown number</th>
<th>Unknown activity</th>
<th>Unknown Number And activity</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1 2 3 4 5</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Am-Be</td>
<td>16</td>
<td>18</td>
<td>7</td>
<td></td>
</tr>
<tr>
<td>Am-241</td>
<td>2</td>
<td></td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>Ra-226</td>
<td></td>
<td></td>
<td>1 (set of needles)</td>
<td></td>
</tr>
<tr>
<td>Ir-192</td>
<td>1</td>
<td>1</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>NDT</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cs-137</td>
<td>2</td>
<td>52</td>
<td>1 (category 4)</td>
<td>3 NDT</td>
</tr>
<tr>
<td></td>
<td></td>
<td>188</td>
<td>1</td>
<td>35</td>
</tr>
<tr>
<td>Cd-105</td>
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<td>9</td>
<td>1</td>
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<td>Kr-85</td>
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<td>Sr-90</td>
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<td>1</td>
<td></td>
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<tr>
<td>Co-60</td>
<td>7</td>
<td>8</td>
<td>50 normal dimension</td>
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<td></td>
<td>79 small discs</td>
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<td>1 set of discs</td>
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<td>Teletherapy source</td>
<td>1</td>
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</table>

3. Comparison between IAEA recommendations and the Egyptian regulations

The IAEA has developed guidelines for Member States to take into consideration as Member States develop their own regulations to control and manage SRSs. These guidelines address both SRSs in-use and unwanted SRSs. To determine the adequacy of current Egyptian regulations, a comparison is being made between IAEA guidelines and the current regulatory infrastructure. IAEA guidelines are taken from two references [6-7] (1) Organization and Implementation of a National Regulatory Infrastructure Governing Protection Against Ionizing Radiation and the Safety of Radiation Sources Interim Report for Comment and (2) Code of Conduct on the Safety and Security of Radioactive Sources. With the comparison between the IAEA and the Egyptian regulations it is noticed that the Egyptian regulations not cover all the required items by the IAEA and obvious shortage especially with the public protection, where the main aim of the regulation that are applied by the Ministry of Health is concerning with the radioactive sources in the medical practices and the medical purposes.

From the tables below we can see that the respective regulatory body in some countries, such as in Egypt, is not effectively independent. This situation is not in line with recommendations provided by the IAEA Code of Conduct [6]. Generally there are two categories of regulatory systems, one with a single regulator, like in the Czech Republic, or one with multiple regulators.

By comparing the IAEA recommendations with the Egyptian regulations it is noticed that the Egyptian regulations do not cover all the required items of the IAEA and there is obvious shortage especially with respect to public protection, where the main aim of the regulation that are applied by the Ministry of Health are concerned with the radioactive sources in the medical practices for medical purposes. There is a clear shortage concerning the control of the other radioactive sources that are operated in other places than hospitals and medical places. Its not clear which the regulatory body is, the Minister of Health or the EAEA, although based on the Law No. 59/1960, which is more powerful than other laws, the main responsibilities are for the Ministry of Health, while the responsibilities of the EAEA are restricted to the unsealed isotopes, the reactors, and to the units that follow. This leads to an obvious confusion about the responsibilities for the management of the radioactive sources and the protection from their hazards. Although there is much confusion, there is no clear system that can resolve the problems that may arise due to the interference between the responsibilities of the bodies in Egypt. According to the current situation, the enforcement of the law and the protection provisions have numerous points of weakness which may lead to violations and accidents. The responsibilities of the licensees, in the Egyptian regulations, are not clear.
Table 2: Comparison between some European countries and the Egyptian regulations [8-9]

<table>
<thead>
<tr>
<th>Country</th>
<th>Regulator</th>
<th>Responsible to</th>
<th>Scope</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>Czech Republic</td>
<td>State Office for Nuclear Safety (SUJB)</td>
<td>Government</td>
<td>Supervision of nuclear facilities. Licensing of activities. Reviewing and approving documentation relating to nuclear safety and radiation protection. Specifying conditions for radiation protection. Specifying emergency preparedness. Maintaining national accountancy/record systems for nuclear materials, licensees, selected import/export items, ionising radiation sources and exposure of the public and personnel. The organisational structure of ERPC includes the following units:  - Environmental monitoring;  - Radiation protection;  - Supervision and inspection;  - Standardisation;  - Information service;  - Commission consisting of representatives from various ministries.</td>
<td>ERPC is the only regulatory body in the field of radiation protection in Estonia. Pursuant to the Radiation Act and the existing subsidiary legislative acts, ERPC is empowered to fulfil the following functions:  - to authorise practices by licensing,  - to inspect practices and sources,  - to monitor and to assess radiation levels,  - to notify about radiation accidents. The Commission is empowered to issue certificates for import, export and transit of strategic goods.</td>
</tr>
<tr>
<td>Estonia</td>
<td>Estonian Radiation Protection Centre (ERPC)</td>
<td>Ministry of Environment.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Hungary</td>
<td>Strategic Goods Import, Export and Transit Control Commission (SGIETCC), HAEA</td>
<td>Ministry of Foreign Affairs.</td>
<td>Co-ordinator in regulatory tasks, supervises the central register of radioactive materials and handles the Central Nuclear Financial Fund. Regulates on radiation protection; SPHAMOS: licensing and supervising tasks related to radioactive materials on behalf of Ministry of Health. Supervision of security. Deal at national level with licensing of all activities connected with the use of radiation sources, co-ordination and control of activities related to the peaceful use of nuclear energy, research and application of nuclear energy in the national economy, safeguards of nuclear materials, immobilisation and storage of radioactive waste (including sealed sources).</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Ministry of Health.</td>
<td>Deals at national level with licensing of all activities connected with the use of radiation sources, co-ordination and control of activities related to the peaceful use of nuclear energy, research and application of nuclear energy in the national economy, safeguards of nuclear materials, immobilisation and storage of radioactive waste (including sealed sources).</td>
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<tr>
<td></td>
<td></td>
<td>Ministry of Interior.</td>
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<td></td>
<td></td>
<td>National Atomic Energy Agency.</td>
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<td></td>
</tr>
<tr>
<td>Poland</td>
<td></td>
<td>Government</td>
<td>The SNSA is responsible for radiological safety at all nuclear Installations and the import, export and transport of radioactive materials. The HIRS is responsible for health and safety of workers and safety in the workplace. HIRS issues licences for the purchase and use of radioactive materials and transport of sources which are used for diagnostics and therapy. The EORP is responsible for licence installation and use of x-ray machines, cyclotrons and SRSs. EORP issues licences for the purchase and use of radioactive materials and transport of sources.</td>
<td>President of NAEA is appointed by the Prime Minister.</td>
</tr>
<tr>
<td></td>
<td>Executive Office for Radiation Protection (EORP)</td>
<td>- Ministry of Health and Population</td>
<td>The NCNSRC is responsible for licensing reactors, opened sources and the protection from ionizing radiation within the EAEA. He EAEA is also responsible for the recovery, storage and disposal of unwanted SRSs.</td>
<td>Director of EORP is appointed by the Minister of Health and Population</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Ministry of Health</td>
<td></td>
<td>Both of the chairman of EAEA and NCNSRC are appointed by the prime Minister of Health.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>and Population</td>
<td></td>
<td></td>
</tr>
<tr>
<td>REGULATORY BODY</td>
<td>EGYPT</td>
<td>POLAND</td>
<td>HUNGARY</td>
<td>CZECH REPUBLIC</td>
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<tr>
<td>Not independent</td>
<td>Not independent</td>
<td>independent</td>
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<th>AEA</th>
<th>MOI</th>
<th>RPC</th>
<th>NSA</th>
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</thead>
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<tr>
<td>MOI</td>
<td>AEA</td>
<td>AEA</td>
<td>&quot;State office for nuclear safety&quot;</td>
<td>SUJB &quot;State office for nuclear safety&quot;</td>
<td>MOI &quot;Radiation Protection Center&quot;</td>
<td>MOH &quot;Nuclear Safety Administration&quot;</td>
</tr>
<tr>
<td>EAEA Decisions</td>
<td>&quot;State office for nuclear safety&quot;</td>
<td>&quot;State office for nuclear safety&quot;</td>
<td>&quot;Strategic Goods Import, Export and Transit Control Commission&quot;</td>
<td>&quot;Health Inspectorate of the Republic&quot;</td>
<td></td>
<td></td>
</tr>
<tr>
<td>EAEA</td>
<td>MOH</td>
<td>MOI</td>
<td>MOI</td>
<td>MOI</td>
<td>MOI</td>
<td>MOI</td>
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</table>

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The penalties for law breaking or the breaking of the protection provisions as mentioned in the Law No: 59/1960 need to have modifications to be suitable for the present economic status and also for the increasing in the quantity and the quality of the threats that may arise from the non-controlled or orphan sources, but this actually needs to new legislation and great efforts. Also, it is very clear the absence of the quality assurance and quality control programs for the management of the radioactive sources. The monitoring and verification processes are not clear in the Egyptian regulations and also there is no technical classification for the areas that involve or nearby the radioactive sources and if there is any monitoring process it will be limited to the instruments that have radioactive sources or for some protection tools and such monitoring is represented by the calibration and investigation processes.

4. Conclusion

With respect to the classification of SSRSs that exist at Inshas storage facilites we can conclude that sources are divided between five categories. Some sources are long-lived radionuclides and do not meet the criteria of waste acceptance for near surface disposal.

The main piece of legislation which governs the regulatory framework in Egypt is the act no. 59 of 1960 relating to the use of ionizing radiation and protection from hazard effects and the act No. 4 of 1994 relating to the environment and the implementing decisions.

REFERENCES


[5] Statistic and Radiation Information Centre, Egypt, (2003), The laws and regulations that arrange the working in the radiation field in Egypt.


Geological disposal of radioactive waste: elements of a safety approach*

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\(^b\) ASN: Autorité de sûreté nucléaire, France
\(^c\) AVN: Association Vincotte Nuclear, Belgium
\(^d\) IRSN : Institut de radioprotection et de sûreté nucléaire, France
\(^e\) ANDRA : Agence Nationale pour la gestion des Déchets Radioactifs, France
\(^f\) ONDRAF/NIRAS: National Agency for Radioactive Waste and Enriched Fissile Materials, Belgium

Abstract. This paper presents the document “Geological Disposal of Radioactive Waste: Elements of a Safety Approach” which is the result of a common undertaking in collaboration in the field of nuclear safety and radiological protection between France and Belgium. The document was elaborated by a working group involving experts from the regulatory authorities, their technical supporting organisations and the implementers. The elements of the safety approach comprise the radiological protection objectives, the safety principles, the safety functions that form basis for the safety case of a geological repository, as well as reflexion on the appraisal of acceptability of the safety case. The document has no regulatory or normative status, nor was the intention to cover all aspects that are relevant for a safety case.

1. Introduction

This paper refers to a document [1] that has been developed within the general framework of the French-Belgian collaboration in the field of nuclear safety and radiological protection. Specific collaboration in the field of the safety approach to disposal in deep geological formations began in June 2000 and led to the creation of a working group involving experts of the regulatory authorities ASN(a) and AFCN/FANC(f), their respective technical support organisations, IRSN(d) and AVN(b), and the implementers ANDRA(c) and ONDRAF/NIRAS(e). This document has no regulatory and/or normative status as well it was not the intention to cover all the aspects of the safety.

Upon initiative of ASN and AFCN/FANC, and with due account for ongoing and published work from ICRP, IAEA and OECD/NEA, regular bilateral discussions have allowed the convergence of ideas and have resulted in the elaboration of the “Elements of a Safety Approach” document. This document establishes a link between the protection of man and the environment (basic objective of protection) and the disposal system through the application of safety principles and the identification of safety functions as well as the development of a framework for the judgement of the acceptability of safety cases by regulatory authorities and decision-makers.

2. Elements of a safety approach

The concepts of safety functions, safety principles and radiological protection principles are described as an integral part of an approach to be taken into account in the development, the implementation and the evaluation of the long-term safety of a radioactive waste repository in deep geological formations. Based on the concentration and containment strategy, the proposed approach provides a structure for and facilitates the judgement of acceptability. In this sense, this proposed safety approach establishes a link between the basic objective of protection and the implementation of the safety principles, the radiological protection principles and the safety functions.

The safety approach is focussed on long-term safety which is purely “passive” in that sense that it cannot be based on maintenance or institutional control with regard to the very long time frames at

* Report of a French Belgian Working Group
The safety principles establish the orientation and methods that provide a framework for the definition of a strategy for developing a repository. The safety functions and their implementation contribute to the establishment of this strategy for the development of a disposal system and the soundness of this strategy is examined in a safety case. Based on these concepts of safety principles and safety functions and taking external programme and design constraints into account, the implementer develops its disposal project, in particular, allocating the safety functions to the different components of the disposal system. This allocation must be supported with arguments and justified in the safety case. The elements of the safety approach presented in this document underline the importance of the qualitative argumentation of a safety case. This allocation means that a strategy where the implementation of the safety functions is consistent with the safety and protection principles must enable the basic objective of protection to be achieved. Figure 1 illustrates the relationship between the different elements of the safety approach.

![Diagram of the elements of the safety approach]

**FIG. 1: Illustration of the elements of the safety approach**

### 2.1. Example of elements of safety approach

As an example of elements of the safety approach, a focus is given in the following paragraphs on the safety principles and the judgement of acceptability.

#### 2.1.1. Safety principles

The important concepts generally used in the development of a safe repository (robustness, passivity, technical feasibility, simplicity) fall under the principle of defence-in-depth and the principle of demonstrability.
When applied to a repository in a geological formation, the principle of defence-in-depth implies the implementation of "multiple safety functions". In this case, it is not the number and redundancy of the barriers as such that take on the greatest importance in terms of safety, but the fact of being able to depend on different mechanisms and/or components to provide safety functions (isolation, containment, limitation and retardation).

The principle of “demonstrability” consists of adopting methods of system development that will make it possible to demonstrate that the functions and performances expected from the repository components will be fulfilled and maintained no matter what reasonably foreseeable disturbance may impact the system.

2.1.2. Judgement of acceptability

The radiological impact is judged on the basis of compliance with the fundamental objective of protection. The bases for the long term radiological assessment have been subject to some developments on the potential weaknesses of the conventional indicators, on complementary elements of these conventional indicators and on scenario classification (normal evolution scenarios, altered evolution scenarios, human intrusion scenarios, “beyond design limit” scenarios and “what-if” scenarios) and associated criteria or reference values. "Conventional" indicators to quantify the radiological impact are effective dose rate and radiological risk.

3. Conclusion

The notions of safety principles, radiological principles, safety functions and the basis of the judgement of acceptability of the radiological impact assessment, which have been discussed in various international co-operative work, have been further integrated in a comprehensive framework for a safety approach. The schematic presentation of a structured safety approach should presumably enable better explanation and communication of the global safety issues to different stakeholders.

Acknowledgement. The document to which this paper is referred has been written by G. Bruno (IRSN), P. De Preter (ONDRAF/NIRAS), A. Grévoz (ANDRA), J. Maudoux (AFCN/FANC), V. Nys (AVN) and P. Rainbault (ASN).

REFERENCE

Human intrusion in near surface disposal
Outcomes of the regulatory meeting held in June 2004 in Brussels

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\textsuperscript{b} IRSN, Institut de Radioprotection et de Sûreté Nucléaire, France
\textsuperscript{c} Environment Agency, United Kingdom
\textsuperscript{d} ASN, Autorité de Sûreté Nucléaire, France
\textsuperscript{e} SSI, Swedish Radiation Protection Authority, Sweden

Abstract. In the framework of Belgian LLW pre-project phase, FANC (Federal Agency for Nuclear Control) and AVN (Association Vincotte Nucléaire) have had many exchanges of views on the following topics concerning the treatment of human intrusion in the assessment of Near Surface Disposal facilities: the regulatory framework and acceptance criteria, the role of the institutional period and the human intrusion scenario(s). During the discussions, the question was raised as to how this matter has been treated in the past for near surface facilities that have already been licensed. This prompted AVN to organise a meeting between the regulatory bodies and their support organizations from some countries having experience with a near surface disposal facility (Finland, France, Spain, Sweden, UK). The objective was to have an open exchange of information and discussion on the aforementioned topics. This regulatory meeting on human intrusion was held in June 2004 in Belgium. This presentation gives an overview of the topics that were discussed and identifies some of the key issues.

1. Introduction

The regulatory framework in relation to Human Intrusion Scenarios for Near Surface Disposal has been the subject of discussions between ONDRAF/NIRAS, FANC and AVN. From these discussions it appeared that if these outcomes should become the basis for regulatory guidance, deeper investigations would be necessary. FANC and AVN considered that the most appropriate way to progress in their investigations, was to organize a meeting with regulatory bodies from other countries that already have near surface disposal facilities. For this reason, a meeting was organised by FANC and AVN with the intention of discussing the following topics:

- The regulatory framework and criteria associated to human intrusion;
- The post-closure period;
- The nature of the scenarios.

Invitations were sent to the regulatory authorities and their technical support organisation of France, Finland, Sweden, Spain, and the United Kingdom. Finland and Spain unfortunately were not able to participate at this meeting but expressed their interest by providing information. France, Sweden and the United Kingdom actively participated. The meeting was hosted by AVN.

2. Human intrusion regulatory framework

2.1. Overview of the current regulatory framework in each country

Each country made a short presentation of its current regulatory framework. The regulatory framework in each country addresses different concerns, depending of the national context. For example, the French BSR 1.2 (Basic Safety rule), requires three types of containment systems including a layered cap on the repository, that on the selected site, natural resources should be limited and that the materials used in construction should not be attractive. BSR also states that the site should be geotechnically stable, located in a non flooding zone, where the hydro-geology is well known and that the activity be limited to allow for a reasonable surveillance period. In the UK, the guidance is deliberately not prescriptive. The use of “best practical means” or qualitative arguments are important, in addition to the quantification of risk. In Sweden, due to the type of facility and its location “the SFR
(the Swedish Final Repository for Radioactive Operational Waste located in a rock cavern under the Baltic Sea), the weight given to passive measures is considerably higher than in France or in the UK. In Belgium, due to the high population density around the site, it is envisaged that the associated cumulative probability for human intrusion is equal to 1 and a dose limit of 1 mSv/y is under consideration for the inadvertent human intruder.

2.2. Main feedback from the workshop

2.2.1. Dose limit and dose constraint

In the application of the dose limit and constraint in the case of human intrusion scenarios, some would argue that a distinction should be made between radiological impacts to the inadvertent intruder and those to the public due to the delayed effects of the intrusion. For the public in general, a dose limit of 1 mSv/y and a dose constraint of 0.3 mSv/y for the radiological impacts due to the delayed effects of the intrusion could be considered. For the inadvertent intruder, it might be argued that the dose constraint should not be applied because, as stated in ICRP 81, §52, the dose constraint does not apply to human intrusion because “… by definition an intrusion event bypasses some or all of the barriers that have been put in place as part of the optimisation of protection”. However, this distinction between the inadvertent intruder and the public is not made in all countries. For example, in the UK, current regulatory guidance makes no such distinction. This topic was the subject of considerable debate at the meeting.

2.2.2. Cumulative probability

During the discussions, participants pointed out that if the cumulative probability reaches “1”, the annual frequency is not necessarily one. This implies that if some individuals belong to the critical group for the normal evolution scenario and to the intruder group, the cumulative effect to them should not be added without careful consideration.

3. Post-closure period

3.1. Overview of the current regulatory framework in each country

In France, it is considered that after the institutional control period, the radiological hazard of the waste should be sufficiently low so that the radiological impact would be acceptable whatever the land-use. The duration of this period is a function of the monitoring and maintenance operation as well as a function of the residual risk. The institutional control period should ensure the following objectives: preventing intrusion, and ensuring a control and a monitoring of the system in order to detect unexpected failure. In UK, the developer may make provision for a period of institutional control and argue that human intrusion is not possible during this period. However, appropriate justification is required. Passive control could provide benefit but credit for this benefit is not always taken in safety assessments. In Belgium, as a result of discussions between FANC and AVN, the institutional control should progressively evolve, ensuring and keeping the same level of safety, from an active control system to a passive control system. Following this evolution, the likelihood of inadvertent human intrusion increases as the efficiency of active measurements diminishes. This should imply that investigations on the consequences of the human intrusion scenarios should be conducted when the safety of the disposal system is only ensured by passive control and surveillance measures. Also, the institutional control period is bounded by a minimum and a maximum duration.

3.2. Main feedback from the workshop

The level of guarding and limitation of access on any site should be adapted to the evolution of the radiological hazard of the waste. A sub-division of the institutional control period in different sub-periods could be determined based on the consequences to intruders at these different time periods. Intermediate criteria like the generic reference levels of intervention recommended by the ICRP could be considered as reference values helping in defining the different sub-periods. During the workshop, the following figure [fig.1] was discussed. The figure offers the basis of a consistent approach for determining the relevance of active and passive measures at different times.
FIG. 1: Illustration of the identification of several sub-periods in the post-closure based on the potential consequences on intruders

The consequences of the human intrusion scenarios at different time periods after the closure of the disposal system allow a link to be defined between the nature of active and passive measures preventing human intrusion and the radiological hazard of the waste content.

“Zone Z3”: The radiological impact is above the high reference value. All unauthorized access (except if authorized) or intrusion should be strictly prohibited.

- “Zone Z2”: The radiological consequences of intrusion are less but still high. If the level of active measures could be reduced, a regular guarding and maintenance of the fence should be still maintained.
- “Zone Z1”: As the radiological impact is now less than the lower value for an intervention, it implies that if unauthorized access or intrusion occurs the consequences for the intruder are reduced. At this time, it could be recommended to seal the gallery and the cave, to dig a trench in order to prevent easy truck access. If the comparisons between the monitoring measurements and the expectations are satisfactory, it could be decided to end monitoring measurements at this time. High passive measures should be already implemented like markers. Memory means should be installed at least at this moment.
- “Zone Z0”: No more active measures should be considered. A free access on site is possible and for any land use the radiological consequences are acceptable.

This way of working recognizes that active measures are a necessity but also recognizes that long-term safety does not solely rely on active measures and the duration of such active measures should be optimized. As the discussed approach addresses the inadvertent intruder, an extension of this approach to the public through the findings of the investigations of the radiological consequences of the delayed effects of human intrusion scenarios, could define a second link between the level of active and passive measures preventing human intrusion and the radiological hazard of the waste content.
4. Scenarios – main feed-backs

It appears through the discussions that the classification of a scenario as a human intrusion scenario is strongly dependent on what has been defined as a barrier in the disposal concept. For example, in France, a scenario considering a well in the aquifer is classified as a human intrusion scenario because the confinement property of geosphere, which is the third barrier, is partly bypassed by the well.

The following categorization of scenarios might be considered: Based on Normal Evolution Scenario (NES), Altered Evolution Scenarios (AES) and Human Intrusion Scenario (HIS). This classification covers as much as possible the different potential evolutions of the disposal system. Firstly the NES should be clearly defined especially by identifying its components, the barriers, the contamination pathways and the outlets to the biosphere. NES scenarios serve as references for the elaboration of AES or HIS scenarios, for monitoring comparison and for the safety margin evaluation. Secondly degradations of the barriers of the repository considered in AES are based on natural events or processes. Finally, HIS scenarios are based on modification or alteration of features considered in the NES due to human activities. Scenarios where the degradations process are due to a combination of human activities and natural events could be elaborated if the degradation processes are independent (cf. wells).

From the discussion, it also appears that Human Intrusion Scenarios are used in determining the radiological impact to the intruder and hence the maximum average activity, and also in determining the maximum allowable waste package heterogeneity. It is expected that different human intrusion scenarios would constrain each of these characteristics. This topic was the subject of considerable debate at the meeting.

Acknowledgement. This paper presents the results of the “Regulatory Meeting on Human Intrusion for Near Surface Disposal Facility”. AVN is grateful to all participants for their active participation.

REFERENCE

Radiation and waste safety in Kenya

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Abstract. Degradation of the environment is occurring at an unexpected rate as a direct or indirect result of man's activities and on a scale that requires international efforts in order to prevent serious long-term negative effects on mankind. To a great extent, the present conditions of accelerating degradation are a result of the intensive exploitation of the earth's natural resources and other human activities in the absence of a clear scientific guidance on an adequate understanding of the consequences. These include ground water pollution due to radioactive waste disposal, discharge of industrial and municipal waste and over-use of pesticides and fertilizers for a long time. The scope of environmental problems is very great and the current topic focuses on the issue of waste disposal not only for its own importance but also as an example of the complex interactions between man, wastes and environment, which could be demonstrated. The objective of radioactive waste management is to deal with radioactive waste in a manner that protects human health and the environment now and in the future without imposing undue burdens on future generations.

1. Introduction
The Republic of Kenya has a land area of 582,650km² and population of about 30.4 million. Kenya became a member state of the Atomic Energy Agency (IAEA) in 1965 and is a party to the non-proliferation treaty (NPT) and signatory to several agreements with the IAEA. The IAEA is one of the UN family organizations that is charged with the mandate of promoting the peaceful uses of radiation and nuclear technology. It sets up regulations, which are enforced by the various competent/regulatory authorities in member states using laws promulgated in these states. The competent authority in Kenya is the Radiation Protection Board, which is established through the Radiation Protection Act, Cap 243 (1985 revised), Laws of Kenya. The functions of the Board are:

- to advise the Minister on matters concerning radiation and waste safety;
- to implement provisions of the Act and related regulations;
- to grant authorizations, as appropriate and impose any necessary conditions;
- to maintain a registry of sources of ionizing radiation.

The Board is empowered by legislation to require notification and grant authorizations, conduct inspections and take enforcement actions. Regulations have been set up which cover exemption, notification and authorization, dose limits, control of medical exposure, radioactive waste safety and transport. These regulations are based upon the published International guidelines.

2. Extent of practices involving sources of radiation
There are 1000 medical diagnostic devices spread among 500 facilities/organizations, 250 nuclear gauges, between 20 and 30 organizations using small quantities of unsealed radioactive materials for research (e.g. radio immunoassay), nuclear medicine and teaching, 80 industrial radiography units spread among 50 user organizations, 7 medical radiotherapy units in 4 hospitals/clinic and one irradiator facility. A second irradiator facility is planned. The sources used in medical therapy are mainly Co-60. A disused Cs-137 unit is also available. Industrial radiography uses mainly Ir-192 sources.

3. Radioactive waste management practices in Kenya
Application of radionuclides is found in industries, hospitals, schools, universities and research institutions. These are mainly sealed sources and to a small extent, unsealed radionuclides. There is no isotope production, research reactor, nuclear power or nuclear fuel facilities in Kenya at present.
Unsealed sources imported annually range from tens to thousands of MBq of H-3, C-14, P-32, S-35, Ca-45, Cr-51, Fe-59, I-131 and In-III. Information is available on the radioactive substances imported since 1986 and work is in progress to compile a national register of radioactive waste in stock or disposed of.

It must be admitted that when new practices are implemented for the first time, nuclear and radiological practices being no exception, more emphasis is placed on what stands to be gained from them than in the waste they will produce. This has given the impression that the waste problem was "discovered" or "recognized" late in the day and that it had been ignored when radioactive substances were first used.

3.1. Radioactive waste

Radioactive waste is any radioactive material that has been, or will be, discarded as being of no further use. It arises from a wide range of activities: the use of radionuclides in hospitals and research laboratories, the use of radioactive materials in industrial processes, production of electricity by nuclear power etc.

Due to the toxicity of radioactive waste, its management must be planned carefully. Waste is concentrated, solidified and placed in containers. Disposal is planned in geological formations, at depths and under conditions, which depends on the content of radioactivity and its decay with time.

![Basic steps in radioactive waste management cycle.](image_url)

Characterization, storage and transportation of waste and materials may take place between and within the basic radioactive waste management steps. The applicability of these steps will vary depending on the types of radioactive waste.

3.2. Waste management practices

In hospitals, solid wastes are stored for decay and are disposed of as inactive waste after monitoring. Excretions of patients (cancer diagnosis and therapy) are discharged into the sewage system.

For research institutions using unsealed sources, liquid effluents pass through a lagoon system with a total residence time of about three months before being discharged into the environment. Incinerators are also used for burning inactive and decayed wastes. Radioactive carcasses are deep-buried or incinerated depending on the levels of activity.

No central facility for treatment, conditioning and interim storage of sealed sources and radioactive waste exists. However, several sealed sources have been conditioned into 200 litre drums at the Material Testing and Research Department, pending final disposal in a repository. In principle, the Department has been appointed as an agency for treatment, conditioning and interim storage of radioactive waste.
3.3. **Waste management regulations**

The Radiation Protection Act regulates the import, manufacturing, possession, handling, export and disposal of any radioactive material including radioactive waste. Two secondary legislations have been elaborated: one covers standards of radiation protection - Legal notice No. 54, based on (ICRP) publication No. 26, while the other covers building and structural requirements and also schedules for licences and fees – Legal Notice No.55.

The Kenya Bureau of Standards has prepared specific Kenya Standard codes of practice on the management of radioactive wastes. They will incorporate principles of management of radioactive waste, limits and conditions for their exemption from regulatory control and technical guidelines for use by waste. Since in 1990, purchasing contracts of sealed sources were required to include a clause assuring the return of the source to the supplier once it has reached the end of its useful lifetime.

3.4. **Proposed radioactive waste management policy**

No clear policy on radioactive waste management exists at the moment. The Regulatory Authority is in the process of formulating it in consultation with the stakeholders. Public understanding is recognized as the key to public acceptance, and communicating to the right audience is the all-important bridge between the two. The social and ethical issues involved and public acceptance of the technical solutions developed for the safe management of RW is important aspects of radioactive waste management and technical outcomes will be in conformity with the policy. The RW management policy will cover the following items:

- Overall objectives, justification and principles of optimization
- Exemption of radioactive waste from regulatory control
- Return of spent fuel sources to supplier
- Obligations of radioactive waste producers
- Central facility for treatment, conditioning and interim storage
- Radioactive waste disposal
- Radioactive waste related research and financial implications.

4. **Way forward**

In order to improve the understanding of problems and processes involved in the disposal of radioactive waste, and for the better assessment of the natural influences of wastes and the environment, the following are recommended:

a) to collect data and experiences (case studies) from existing, well-controlled waste disposal facilities;

b) to encourage research aimed at better understanding for the nature, behaviour and characteristics of different types of waste in order to better classify the wastes;
c) to encourage scientific investigations on the long-term behaviour of natural barriers to waste movement, including sealing materials such as clay layers;
d) to establish and promulgate recommendations and guidelines for
   i) safety concepts in waste disposal,
   ii) site selection criteria,
   iii) disposal performance criteria, and
   iv) the needs of earth-science data for site selection.

The transfer of research results experience, and methodologies from the highly advanced field of nuclear waste disposal should be promoted for application to all aspects of toxic-waste disposal, for which much less reliable geo-technical information now exists. All waste disposal operations should be preceded by comprehensive assessment of their environmental and health impacts. Rational choices on toxic waste disposal should be based on cost/benefit analysis. It should be noted that decision makers and beneficiaries (of the practices that generate waste) are current generations while it is future generations, which will have to face the risks. One of the cardinal principles of radioactive waste management is to minimise the production of waste and manage the waste produced so as not to impose undue burdens on future generations.

REFERENCES

Regional repositories – The best alternative for countries with unfavorable conditions for siting their own disposal sites. The Latin American Case.


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Abstract. The use of radioactive materials and radiation sources as well as the production of radioisotopes and labeled compounds may always produce radioactive wastes which require adequate management and, in the end, disposal. However, there are countries in the Latin American region whose radioactive waste volumes do not easily justify a national repository. These countries would benefit from regional co-operation for the disposal. The intentions of this paper are to discuss the advantages of having a common repository (repositories) in Latin American Region and to promote the cooperation between countries interested in advancing the concept of shared storage and disposal. Small potential users of a common facility should be able to speak with a common voice when discussing the options with international bodies or with potential host countries offering such back-end services.

1. Introduction
Several Latin American (LA) countries have been firmly committed to the peaceful applications of ionizing radiation in medicine, industry, agriculture and research in order to achieve socioeconomic development in diverse sectors. Consequently, the use of radioactive materials and radiation sources as well as the production of radioisotopes and labeled compounds may always produce radioactive wastes which require adequate management and, in the end, disposal. Some countries with small nuclear programmes may not have the resources or the full range of expertise to build their own repositories for long lived radioactive wastes. The amount of waste generated is also very small comparing with countries operating nuclear reactors. For these countries it is sensible to consider an option of shared deep disposal facilities [1].

2. Main waste inventories and storage conditions
At present, small volumes of long-lived radioactive waste are stored at the few centralized installations and in many cases at user’s premises. The main inventories of RadWastes stored in LA countries are Disused Sealed Radioactive Sources (DSRS). Storage conditions vary, which imply poor security and a risk for misuse of these materials. Some examples of storage facilities from six LA countries are shown in Fig.1.

3. Final disposal?
The disposal of the longer-lived and higher activity DSRS in a national deep geological repository is clearly a sensible solution for countries that are developing such facilities. However, many other countries with no nuclear power programme will not have a deep repository. An alternative currently being evaluated is the use of properly designed and managed borehole disposal facilities [2]. This technology may be suited for providing a safe and affordable disposal route in developing countries. The development of regional or multinational facilities would enable a number of countries, each with only a small inventory of spent sources, to share a repository or a borehole disposal facility situated in one volunteer state. A larger country with significant volumes of waste requiring deep disposal might agree to help developing countries [3]. The most typical examples are the already conditioned disused
226Ra sealed sources, as well as 241Am and neutron sources (Am/Be, Pu/Be) stored in different places awaiting disposal, but where? It looks like some countries are keeping open the option of disposal in a regional framework. Several LA countries generate such small volumes of long-lived wastes that it would be economically unreasonable to attempt final disposal in these countries. Cost for site selection, site characterization, design, construction, operation, closure and licensing of the repository is totally disproportionate with regards to the size of the nuclear programmes. Many LA countries have not established a national centralized strategy for collection, transportation, handling, treatment, conditioning, storage and disposal of radioactive waste. There are absolutely no plans for the final disposal of long-lived radionuclides. The only options for managing high specific activity or long-lived disused radiation sources are to store them indefinitely. But indefinite storage is not a sustainable solution. On the other hand there are insufficient technical capabilities to implement a deep repository.

4. Regional repository/repositories

Different approaches for implementing multinational repositories are been distinguished. This paper focuses on the “bottom-up” initiative expressed in [4] for countries with a common problem that they cannot easily solve alone coming together to explore common solutions. Several LA countries are small in area, with very limited nuclear power programmes and hence waste inventories, including wastes containing long-lived radionuclides.

On the other hand, some national geological repositories are planned to be implemented (for spent fuel and high level waste) in the American continent, in countries able to demonstrate an adequate level of technological skills, resources and nuclear safety infrastructure. The basic issues involved in a regional repository are not much different from those related to national projects. Such regional repositories, which would build upon the best international practices in radioactive waste management, would reduce the number of location at which radioactive materials are disposed of, and then will contribute to a global cleaner environment.

4.1. Examples and initiatives of multinational co-operation

Within USA some DOE sites dispose of LLW (Handford, Idaho, Los Alamos, Nevada Test Site, Oak Ridge and Savannah River). Two of these (Handford and Nevada Test Site) have historically served as regional LLW disposal sites [5]. Another example is explained in [6] in which six States worked together to provide for the disposal of regionally generated Low level RadWastes. To accomplish this, the six States (Indiana, Iowa, Minnesota, Missouri, Ohio and Wisconsin) formed the Midwest Interstate LL RadWaste Compact.

There are several examples of international or bilateral agreements in waste management, in which commercial organizations in some countries accepted responsibility and custody of waste generated in other countries. One example is the sharing of processing and conditioning facilities for radioactive waste as well as reprocessing facilities for nuclear spent fuel on a multinational basis. Other examples are the return of US spent research reactor fuel to the USA and the return to the former USSR of commercial spent fuel of USSR origin. In the recent past, a wish or a need for multinational cooperation has been expressed by a number of countries, which are not in a favorable position to implement self sufficiently national repository programmes for all types of waste arising in their countries and/or which seek to benefit from multinational cooperation for the implementation of a nuclear repository [7].

The topic of the common Czech-Slovak deep geological repository had been regularly discussed on the Czech-Slovak seminars by DECOM – Slovakia and NAVRA – Czech Republic. Two years ago, DECOM Slovakia took a more proactive approach to the development of a regional deep geological repository for the European countries [1].

The Pangea Project also focused on this topic. It was based on a particular “high isolation” concept and various regions of the world possessing specifically favorable geological and geographical environments were identified in Australia, Southern Africa, Argentina and China [8]. A commercial approach to implement an international repository was developed in detail.
ARIUS (Association for Regional and International Underground Storage), is a small group of organizations from eight countries cooperating in an association to support the concept of sharing facilities for storage and disposal of all types of long lived radioactive wastes, [8, 9].

The SAPIERR project (SAPIERR = Support Action on Pilot Investigations on European Regional Repositories) is managed by organizations based in Slovakia and Switzerland and has the support of 21 organizations from 14 countries [10]. The project aims to bring together countries with an interest in investigating the possibilities for shared repositories for spent nuclear fuel and high-level RadWaste.


Another example is the transfers and exchanges of hazardous waste generated in chemical and nonferrous industries. These well-implemented commercial practices aim at the optimal use of disposal opportunities in certain countries and are regulated by The Basel Convention.

4.2. International organization’s criteria

Although the responsibility for the safe management of radioactive waste remains fully on national institutions, a broad consensus exists of its international dimension. Involvement of international organizations with regional or worldwide domains may enhance and accelerate implementation of a multinational facility [12]. The potential advantages of the concept were pointed out by various institutions, including the IAEA, European Community (EC), US National Academies and the World Nuclear Association (WNA). The IAEA, has prepared and published some technical documents [13] and has talked publicly on the advantages of multinational repositories [14, 15]. The European Commission has also recognized the potentially beneficial role of shared repositories.

4.3. Benefits and challenges

As stated in Ref. [12] important benefits and challenges for countries contemplating a multinational repository programme are in areas related to safety, security, non-proliferation, economics, institutional requirements, and public acceptance and support.

It is relatively easy to identify countries that could benefit most directly from the availability of shared waste repositories. These are the countries that will have difficulties in establishing an expensive and complex disposal facility for small quantities of wastes.

National advantages for the host country: There are numerous ways in which a country satisfying relevant requirements for implementing the regional repository could benefit from hosting a shared such facility. There is an obvious benefit of direct financial compensation.

A hosting country would increase its international influence, since it would become an important player on the global nuclear security and the non-proliferation activities.

Regional (Global) advantages: There are also global advantages of having disposal facilities available to all countries where radioactive materials are to be found - irrespective of their size and economic status. These global advantages are of an environmental nature, but also directly related to nuclear safety and security. This aspect has recently grown enormously in importance due to the threat of misusing such material by terrorist groups. It is definitively safer and more secure approach by gathering sensitive radioactive and nuclear materials into fewer, well-guarded sites.

Concerns: The concern is sometimes expressed in large countries that such support could impact negatively on their national programmes, since their own population might assume that this could lead to waste import. This concern can be alleviated by a firm statement of a strictly national policy, anchored if necessary, by appropriate legislation.

As mentioned, there are insufficient technical capabilities to implement national deep repositories in each country. Someone could say that direct assistance, help, expertise, knowledge and funding could be sent to smaller nations. That is definitively truth, but the practical experience has demonstrated that unfortunately this idea does not materialize in many cases. Let’s assume that this assistance and
funding will be possible, and then more than 30 repositories would be developed in the LA region as shown in Figure 2. Does this looks environmentally acceptable? The majority of these extremely expensive repositories would be constructed only for a very few cubic meters of long-lived wastes (mainly DSRS).

4.4. Multinational co-operation in Latin America

There are some examples of regional co-operation through the IAEA in wide range of research, medical and industrial applications of ionizing radiation within the LA region [16]. Closer related with management of RadWaste in the LA region an IAEA regional project was carried out for defining a regional strategy for managing research reactors spent fuel. This project proposed solutions that are technologically and economically suited to the realities in countries involved [17].

Recently (May 2005) a regional workshop for planning of waste management activities on the Latin American Region and the Caribbean region, organized by the IAEA, was held in Santiago, Chile. Representatives from 8 countries analyzed the current situation of RadWaste management in the region and established a common approach to resolve the main difficulties in this field.

It is important to define the countries that have implemented a national disposal solution, but which openly and actively support the communal efforts of others in need of shared disposal facilities. It would be beneficial to reach the agreement by large countries in the region with suitable repository sites, to transfer these to a regional site that can operate a disposal facility available to Latin America. As mentioned in WM’02 [18], our challenge is negotiating a win-win situation for all parties involved.

Our efforts should be focused, at present, in convincing as much organizations as possible on the necessity and technical viability of the regional repository.

FIG. 1: Storage of disused sealed radioactive sources and radioactive Wastes in six Latin American Countries.
5. The way forward

A LA regional repository can enhance the regional safety and security by making timely disposal options available to a wider range of countries. For several LA countries, regional repositories are a necessity, if safe and secure final disposal of long-lived radioactive waste and disused sealed sources is to replace indefinite storage in surface facilities.

There is a clear need to achieve sufficient consensus on the necessity for implementing a regional repository in Latin America for the final disposal of very small amount of long-lived radioactive wastes from LA countries. This is a technical and moral responsibility of the present nuclear community. The concept of regional repositories would appear to make a good sense for our case and seems to be the best alternative in our region.

It is necessary to continue the identification of all countries in the region that are openly interested in being potential users of a common facility; and the ones that are prepared to consider the possibility of hosting a shared facility.

Closer co-operation with the IAEA and other well-recognized organizations involved in this area (such as ARIUS, DECOM-Slovakia) would be very beneficial to promote and develop the necessary and technically viable regional solution in Latin America.

The paper described the main ideas of a group of specialists who recognize that the credibility of such regional project would also be greatly enhanced if other Latin American potential user-countries were prepared to be identified and to enter into serious discussions.
REFERENCES


Regulatory framework for radioactive waste management in Nigeria

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Abstract. A Nigerian Radioactive Waste Management Policy and Strategy document has been drafted, presenting the national commitment to address the country's radioactive waste management issues in a co-ordinated and co-operative manner. The policy, regulatory framework, legislation, regulations and the strategy concerning the management of radioactive wastes in Nigeria are described in this paper.

1. Introduction

A Nigerian Radioactive Waste Management Policy and Strategy document has been drafted and it presents the national commitment to address the country's radioactive waste management issues in a co-ordinated and co-operative manner. The scope of this policy relates to all radioactive wastes, including operational radioactive liquid and gaseous effluent (waste discharges). The Nuclear Safety and Radiation Protection Act empowers the Nigerian Nuclear Regulatory authority to license the disposal of any radioactive material, nuclear material and radioactive waste. As a strategy, the policy recommends the setting-up a National Committee on Radioactive Waste Management and the establishment of four Designated Waste Management Facilities to be licensed by the Authority. The establishment of a Radioactive Waste Management Fund is also recommended in the policy document. The vehicle for the implementation of the policy is the Nigeria Radioactive Waste Management Regulations to deal with the three broad groups of industrial, medical and research applications. These have also been drafted and they provide for the setting up of the basic technical and organizational requirements to be complied with by the waste generators and operators of waste management facilities in order to ensure the protection of human health and the environment from the hazards associated with radioactive waste within and beyond national boarders, at present and in future. The regulations cover the requirements associated with such steps in waste management as collection, segregation, characterization, treatment, conditioning, storage and preparation for transport of radioactive waste arising from medical, industrial and research facilities where radioactive materials and sources of ionizing radiation are produced, used or handled. The waste disposal options being considered are near surface facilities and the borehole disposal concept.

2. Policy and regulatory framework

The Nigerian Radioactive Waste Management Policy and Strategy serves as a national commitment to address the country's radioactive waste issues in a co-ordinated and co-operative manner. It envisages that the management of radioactive waste in Nigeria shall be in accordance with national objectives and recognized international principles as set out in the Policy. The scope of this policy relates to all radioactive wastes, including operational radioactive liquid and gaseous effluent (waste discharges), which is permitted to be released to the environment routinely under the authority of the Nigerian Nuclear Regulatory Authority.

The international community through the International Atomic Energy Agency (IAEA) has developed a comprehensive set of principles for the safe management of radioactive waste. These basic principles are applicable to all countries and can be applied to all types of radioactive waste, regardless of its physical and chemical characteristics or origin. As a member state of the IAEA therefore, and in accordance with National and International objectives, it is the Nigerian Government's policy to deal with radioactive waste in a manner that protects human health and the environment, now and in the future. The policy principles also provides for the management of radioactive waste in such a way that
predicted impacts on health of future generations will not be greater than relevant levels of impact that are acceptable today and that no undue burdens are imposed on the future generations. The policy also states that radioactive waste shall be managed within appropriate national legal framework with clear allocation of responsibilities and the provision of independent regulatory functions.

The stated policy objectives are in consonance with the national objective of sustainable development, which is development that meets the needs of the present without compromising the ability of future generations to meet their own needs. In addition to the internationally accepted principles, waste management in Nigeria shall be managed in accordance with the principles that the generator of radioactive waste is responsible for the waste generated by him and that the financial burden for the management of radioactive waste shall be borne by the generator of that waste. All radioactive waste management activities shall be conducted in an open and transparent manner and the public shall have access to information regarding waste management where this does not infringe on the security of radioactive material and decision-making shall be based on proven scientific information and recommendation of competent national and international institutions dealing with radioactive waste management. Other objectives include the Precautionary principle that is where there is uncertainty about the safety of an activity a conservative approach shall be adopted and that Nigeria will not import radioactive waste.

3. Legislation and regulations

The Nigerian Nuclear Regulatory Authority is established by law [1] with the responsibility for nuclear safety and radiological protection regulation in Nigeria and amongst its functions are the regulation of possession and application of radioactive substances and ensuring the protection of life, health, property and the environment from the harmful effects of ionising radiation. The law also empowers NNRA to make regulations governing the nuclear safety and radiological protection in Nigeria.

In exercise of this powers the NNRA drafted the Nigeria Radioactive Waste Management Regulations with the objective of setting up the basic technical and organisational requirements to be complied with by the waste generators and operators of waste management facilities in order to ensure the protection of human health and the environment from the hazards associated with radioactive waste within and beyond Nigeria’s boarders, at present and in future.

The Regulations covers the requirements associated with such steps in waste management as collection, segregation, characterisation, treatment, conditioning, storage and preparation for transport of radioactive waste arising from medical, industrial and research facilities where radioactive materials and sources of ionising radiation are produced, used or handled. Technologically Enhanced Naturally Occurring Radioactive Materials (TE-NORM) waste is not covered by these regulations.

The application of these Regulations shall be in addition to the Nigeria Basic Ionising Radiation Regulations 2003 (NiBIRR) and the Nigeria Technologically Enhanced Naturally Occurring Radioactive Materials Regulations and any other existing ionising radiation and nuclear regulations as well as any transport regulations in force at the commencement of these regulations.

The regulations places primary responsibility for the safe management of radioactive waste with the waste generator who shall take all necessary actions to ensure the safety of radioactive waste unless the responsibility has been transferred to another person or organisation as approved by the Authority.

The Authority is responsible for enforcement of compliance of the provisions of these Regulations and all other relevant requirements by waste generators and the Operators of Designated Radioactive Waste Management Facilities as established under the Regulations and the implementation of the licensing process for generation and management of radioactive waste.

The waste generator is responsible for on-site segregation, collection, characterisation, and temporary storage of the radioactive waste arising from his activities and discharge of exempt waste. All radioactive waste that are not expected to decay to clearance levels within one year from the time of its generation shall be transferred from the waste generator to the Designated Radioactive Waste Management Facilities.
The Designated Radioactive Waste Management Facilities are designated to serve as the centres for collection and transportation of all radioactive waste from the waste generator’s establishments and for treating, conditioning and storing the radioactive waste requiring more than one year decay period to bring down the activity level to below clearance levels. Designated Radioactive Waste Management Facilities have the responsibility to discharge exempt waste and to store conditioned radioactive waste until a disposal facility is established and becomes operational and the waste has been disposed of, or the waste has been transported abroad for further processing and disposal.

4. **Strategy**

NNRA is establishing National Committee on Radioactive Waste Management, for Designated Radioactive Waste Management Facilities in the country as well as the Establishment of Radioactive Waste Management Fund. The NNRA is taking advantage of regional and international workshops, training courses and seminars to build up capacity in radioactive waste management both for regulators and generators/managers of radioactive waste management facilities.

Three possible geological formations for near-surface and three others for deep geological repository have been identified. It is recognised that the characterisation and eventual site for each type of repository involves a long process study and safety assessment. The NNRA is initiating actions on the further study of these sites in addition, the NNRA is to investigate the types of repository to be studied.

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The status and trends in the area of conditioning, storage and disposal of radioactive waste in the Russian Federation

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Abstract. Presently the main goals in the area of radioactive waste (RW) management at Nuclear Power Plants (NPPs) and radiochemical plants in Russia are introduction of conditioning facilities for all generated and accumulated RW and construction of RW disposal facilities. The measures planned aim to improve the existing RW management infrastructure by means of adoption of a federal Law on RW management and creation of a centralized national system of RW management. The investigations linked with geological repositories construction are carried out in Siberia now. As a result of the geological research carried out in the Krasnoyarsk region in the 1990s, the territory of Nizhnekanisky massif was recognized as potentially suitable for the construction of a disposal facility for solidified RW of the Mountain Chemical Combine (MCC) and untreated Spent Nuclear Fuel (SNF) of NPPs. The project of creation of an International Center for SNF management in Russia is discussed.

1. Introduction

The following basic aspects are linked with the status and trends in the area of RW management in the Russian Federation now:

- Intention to realize in the Russian Federation the conception of closed nuclear fuel cycle (NFC) based on power water reactors and fast reactors. This conception includes SNF reprocessing, high level waste (HLW) partitioning, long-lived RW disposal in a geological repository.

- Further extension of nuclear power industry by means of commissioning of new NPPs (in particular for the period till 2010 of new NPP reactors with the summary electric capacity of 3000 MWt).

- Gradual equipping of NPPs with conditioning facilities.

- Intensification of measures and works aimed at the solution of ecological problems of Production Association “Mayak” (PA “Mayak”) including discontinuation of use and closure of liquid RW open storage ponds.

- Intention to improve the existing national RW management system.

- Official interest to consider the project of SNF management International Centre construction in the Russian Federation.

2. The regulatory and legal framework for RW management and its development in Russia

At present, activities in the area of RW management are governed by a number of federal laws as well as by standards dealing with certain technological aspects. In general current regulatory and legal framework is sufficient for ensuring the activities of enterprises where RW is generated, processed and stored. Further development of the regulatory and legal framework for RW management is connected with the following issues:

(a) development and adoption of a federal law on RW management in Russia stipulating the key organizational and financial aspects.

The purpose of this law is a complex legal solution of the RW management problem, consolidation of provisions now contained in a dozen of laws, decrees and regulations in one law, regulation of the RW management economic mechanism.
(b) the necessity to regulate the existing legal framework with regard to the federal industry-wide law on technical regulation.

The purpose of the technical regulation law is to promote development of market relations in Russia by removing ungrounded restrictions in business. The law is based on a democratic principle - "Everything is permitted which is not prohibited". The law provisions require the development of technological regulations in all branches of industries including regulations for the nuclear industry, based on this principle.

The prevailing principle of the current nuclear law is quite different - "Everything is prohibited which is not allowed". The essence of this principle is to exclude the possibility of nuclear and radiation incidents and accidents by introducing a wide range of grounded restrictions at all stages of radioactive materials management.

Thus, the basic principles of the current nuclear laws and the industry-wide law on technical regulation contradict each other. In this connection work is under way now to find a compromise allowing the use of both principles for adjusting the current legal documents and working out new ones, including those dealing with RW.

3. The Infrastructure of the Russian RW management system and its development

The RW management system of NFC enterprises is based on the following conditions:
- Each enterprise (NPP, radiochemical plants and others) is responsible for collection, treatment and storage of their own RW.
- Funding of RW activities is based on the resources of various programs (federal, agencies and others – for big projects) and resources of enterprises (for current works).

It is supposed to improve the existing national RW management infrastructure in the near years by means of the following measures:
(1) Creation of a national Fund for financing the end stages of RW management (including construction of long-storage and disposal facilities).
(2) The creation of a centralized national system (including managing corporation) responsible for transportation of conditioned RW from enterprises and operation of long-storage and disposal facilities. Presently a set of the documents necessary for realization of these measures is being prepared.

4. Radioactive waste management at NPPs

Presently 16 reactors of VVER (PWR-type), 10 reactors of RBMK (BWR-type), 5 reactors of other types with the summary electric capacity of ~22 000 MWT are in operation in Russia. The history of evolution of RW management system at Russian NPPs [1] may be divided into 3 stages.

4.1. The first stage (NPPs commissioned before the 1970s)

At the first Russian NPPs the RW management was based on the provision that the RW generated at NPPs is to be conditioned when the NPP is decommissioned. In connection with this provision solid RW was placed in storage facilities without conditioning. Liquid RW was processed by evaporation and sorption. The evaporator bottoms and pulps of spent sorbents were directed for storage into steel tanks.

Long-term storage of liquid RW concentrates and unconditioned solid RW contradicts modern principles of ecological security, and this led to the abandonment of this technology. Unfortunately, a major problem, which remains unsolved, is the necessity to extract large quantities of liquid concentrates and solid RW from storage facilities and subsequently condition them. At present storage facilities of NPPs hold ~130 000 m³ of bottoms and pulps and ~150 000 m³ of solid RW.
4.2. The second stage (from the 1970s up to now)

Projects of new NPPs provide for obligatory conditioning of liquid and solid RW. At the Leningrad and Kalinin NPPs bituminization facilities are in operation. Incineration and compaction facilities were introduced at a number of NPPs. In 2005 cementation facility started operation at the Volgodonsk NPP.

Presently repositories of NPPs RW are not available in Russia that is why all RW is stored at NPPs. The type of storage facility depends on the technologies applied at the NPP. Bottoms and pulps are stored in tanks, solid RW and bitumen compound – in concrete cells, salt cake from deep evaporation of liquid RW – in containers.

NPPs carry out measures aimed at reducing the original RW volumes, scientific research institutes work at reducing the volumes of conditioned RW, primarily by way of extracting the major part of non-radioactive components from RW during the conditioning process.

4.3. The third stage (the next 10-15 years)

The main goals are as follows:
- to equip all NPPs with the necessary set of facilities for conditioning of stored and currently generated RW with packaging the end product into containers. Two solidification methods will be used for conditioning liquid RW – bituminization (at the Leningrad and Kalinin NPPs) and cementation - at other NPPs. It is planned to give up the practice of placing the bitumen compound into cells and pass to its placing into 200 l drums;
- to introduce technologies of bottoms selective purification from radionuclides. The aim of this operation is to remove from RW the major part of non-radioactive components, thus drastically reducing the quantities of conditioned RW sent for long-term storage or disposal;
- as and when NPPs are equipped with conditioning facilities it is planned to pass to the container type storage of conditioned RW in concrete 1,5 m³ containers;
- to build container-type storage facilities at NPPs;
- to build regional NPPs RW repositories. At present some sites potentially suitable for repositories construction are under investigation in Russia. Such sites were found in the vicinity of the Kalinin, Leningrad and Kola NPPs.

5. Radioactive waste management at radiochemical combines

There are three radiochemical combines operating in Russia now: MCC and Siberian Chemical Combine (SCC) in Siberia, and PA ‘Mayak” in the Urals. RW management systems at these enterprises are characterized by specific features.

5.1. Radioactive waste management at MCC and SCC

Liquid HLW is stored in tanks [2]. In future all liquid HLW should be immobilized in stable matrices and sent for long time storage and disposal.

For more than 30 years liquid low level waste (LLW) and intermediate level waste (ILW) of MCC and SCC were disposed by injection into isolated underground horizons at the depth of 180 to 500 m. At the sites of underground liquid low and intermediate level waste (LILW) disposal a ramified system of watch boreholes is available which allows the constant monitoring of radiation and hydrogeologic situation. The monitoring results provide evidence that for the forecast period of time (thousands of years) the injected solutions will remain located at the sites of their disposal, hence this way of LILW disposal may be considered environmentally safe.

The general information on volume of disposed liquid LILW is shown in Table 1.
5.2. Radioactive waste management at PA “Mayak”

Liquid HLW, all currently generated and a part of accumulated in tanks, is solidified by vitrification \([3,4]\).

The system of LILW management of PA “Mayak” is based on the use of special open type ponds for collection and storage of liquid waste. The location of ponds in the total scheme of water supply at PA “Mayak” is shown in fig. 1. The characteristics of these ponds are given in Table 2.

The use of special storage ponds for long-time storage of LILW at PA “Mayak” does not correspond to the modern environmental safety requirements. For this reason in 2003 the Ministry of Atomic Energy of the Russian Federation approved a Program of improving the ecological situation at PA “Mayak” for the period of 2003-2025. In the field of LRW management the Program provides for the following issues:

- introduction of modern LILW conditioning technologies and construction of a near surfaces storage facility for cement compound;
- use of the vitrification technology for solidification of the main part of ILW;
- discontinuation of discharging LILW into special storage ponds and, primarily, into V-9 (Karachay), where works are under way to decrease the water area and to immobilize bottom sediments;
- closure of special storage ponds V-9 and V-17.

Taking into account the foregoing, PA “Mayak” together with scientific and design institutes has developed a set of technological schemes and equipment.
6. Main research and development in the area of RW management

The following main research and development in the area of RW management in Russia should be noted:

- improvement of vitrification facility of the EP-500 type on the basis of ceramic melters;
- development of new host matrices for immobilization of HLW fractions;
- development of a vitrification technology for liquid ILW (separately and together with HLW) and pulps;
- development and testing of a pilot vitrification facility with calcinator and cold crucible type melter for HLW- fractions immobilization;
- testing of LILW treatment and conditioning technologies at PA “Mayak”;
- testing of selective sorbtion technology of bottoms cleaning of till non-radioactive condition at some NPPs;
- geological and other researches of sites potentially suitable for construction of underground disposal facilities for RW containing long-lived radionuclides and untreated SNF.

7. Activities in the area of geological disposal [5-9]

7.1. Predisposal of HLW

Ceramic melter is used to vitrify the HLW now and will be used in the future. Presently the end product of vitrification is sodium-aluminiumphosphate glass. In future it can be replaced by borsilicate glass. The results of the EP-500 vitrification facilities used at PA “Mayak” are presented in Table 3.

Table 3. The use of the EP-500 vitrification facilities at PA”Mayak”

<table>
<thead>
<tr>
<th>Vitrification facilities</th>
<th>Operation duration, years</th>
<th>Liquid HLW solidified, m³</th>
<th>Glass produced, m³</th>
<th>Activity included in glass, million Ci</th>
</tr>
</thead>
<tbody>
<tr>
<td>EP-500/2</td>
<td>1986-87</td>
<td>1000</td>
<td>160</td>
<td>3.9</td>
</tr>
<tr>
<td>EP-500/3</td>
<td>2001- till now</td>
<td>7200</td>
<td>1560</td>
<td>163</td>
</tr>
<tr>
<td>EP-500/4</td>
<td>2006</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>EP-500/5-6</td>
<td>~2010</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
This glass will be placed in the near surface storage facility at PA "Mayak" pending the construction of a geological disposal.

According to the RT-1 radiochemical plant modernization plan, improved SNF reprocessing technology will produce some fractions of HLW, including fraction of long-lived actinides. It is planned to immobilize actinide elements fraction in the mineral-like matrices characterized by high geochemical stability that nearly equal to stability of natural materials.

Presently some technologies for synthesis of mineral-like matrices are developed and tested. Use of cold crucible type melter for immobilization of fraction of long-lived actinides seems the most promising. This facility permits to carry out high temperature matrices synthesis and to produce ceramic materials, in particular, Synrock.

7.2. **The situation in the area of geological repositories construction**

The investigations linked to geological repositories construction were started in Russia more than 40 years ago. Some potentially suitable underground massifs were investigated including salt, clay and granite.

The two suitable sites can be considered for HLW geological repositories construction now. The first one - in the Urals for PA"Mayak” HLW, the second one - in Siberia for MCC HLW and for untreated SNF disposal as well.

7.2.1. *The situation with PA “Mayak” HLW disposal*

A search for suitable sites for HLW of PA “Mayak” disposal has been carried out since the 70s. The near-surface zone has been investigated in detail up to 100 m depth, in some places – up to 1200 m. The entire geological investigations have been carried out on the basis of 18 boreholes 200-1200 m deep each. The results of these researches make it possible to conclude that igneous rock massif is potentially suitable for disposal of RW containing long-lived radionuclides. The main task for the next period is to complete main geological, hydrogeological and geophysical investigations of this site and to prepare design documents for underground laboratory construction. Unfortunately these researches are no longer properly financed.

7.2.2. *The situation with MCC HLW disposal*

As a result of the geological researches carried out in the Krasnoyarsk region in the 90s, the territory of Nizhnekansky massif was recognized as potentially suitable for construction of disposal for solidified RW of the MCC, RW of future radiochemical plant RT-2, SNF of NPPs with RBMK-reactor. Currently researches are carried out at the “Yeniseisky” site, located 7 km from MCC. A subsite in the southwest part of the “Yeniseisky” site has been recommended for further study. The results of geological and hydrogeological investigations and preliminary calculations have shown a potential possibility of creation of a RW underground disposal facility in the southwest part of the “Yeniseisky” site at a depth of more than 700 m. At the first stage it is planned to construct an underground laboratory.

8. **Considerations concerning the creation of an International SNF Management Centre in Russia**

Now that laws and regulations have been amended, Russia has a legal possibility to accept on a commercial basis for storage and processing foreign SNF. It is expected, that the storage facilities for SNF at MCC may be used for realization of such projects. These storage facilities are the first stage of the new radio-chemical plant RT-2. In the recent years on the initiative of the IAEA the question of creation of an International Centre for SNF storage [10] has been studied in detail. President V.V. Putin at the meeting with Mr. M. El-Baradei, Director General of the IAEA expressed his support to this idea in principle. Two sites for the International Centre can be considered as potentially acceptable: there are the above mentioned Yeniseisky-site in the Krasnoyarsk region and a site near the Priargun Chemical Mining Combine in the East Siberia. This combine is situated in a sparsely populated area, has highly trained personnel and modern transport communications. Geological prospecting carried out
earlier showed that several sites can be accepted for creation of a geological storage facility in the rock mass.

The project of creation of an International Centre for SNF management is to be further discussed and is likely to lead to practical decisions.

9. Summary

1. Presently the main goals in the area of RW management at NPPs and radiochemical plants in Russia are introduction of conditioning facilities for all generated and accumulated RW and construction of RW disposal facilities.

2. The measures are planned aimed to improve the existing RW management infrastructure by means of a federal Law on RW management adoption and a centralized national system of RW management creation.

3. The investigations linked to geological repositories construction are carried out in Siberia now. As a result of the geological researches carried out in the Krasnoyarsk region in the 90s, the territory of Nizhnekansky massif was recognized as potentially suitable for construction of disposal facility for solidified RW of the MCC and untreated SNF.

4. The project of creation of an International Centre for SNF management in Russia is discussed now.

REFERENCES


Artificial radionuclides in the ecosystems of the gulfs in the Kola peninsula and in the areas of radioactive wastes keeping

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Abstract. The results of radioactive contamination investigations carried out in the Kola Peninsula gulfs’ ecosystems in the vicinity of the places of radioactive wastes keeping are presented. These investigations were conducted during 1994-2004 by specialists of the Murmansk Marine Biological Institute KSC RAS. The data on the artificial radionuclides contents in bottom sediments, water and hydrobiontes of the Kola, the Motovsky Bays and gulfs of the Barents Sea are presented. Coefficients values of caesium radionuclides distribution between water and bottom sediments of the Kola Bay obtained experimentally and developed on the basis of the data on the composition and amount of artificial radionuclides in bottom sediments classification of bays and gulfs are also given.

1. Introduction
On the coastal areas of the bays and gulfs of the Kola Peninsula a large number of radioactively dangerous objects, such as places of basing military and civil vessels with atomic energy power installations, repairing and technological enterprises, the places of temporally kept spent nuclear fuel and solid radioactive wastes are located. Since 1994 the Murmansk Marine Biological Institute KSC RAS has been investigating distribution of the artificial radionuclides in the bays and gulfs of the Barents Sea.

2. Anthropogenic radionuclides in the bays and gulfs of the Barents Sea
Bays Pechenga, Teriberka and Yarnyshnaya are characterized by lack of 60Co and low contents (1-13 Bq/kg) of 137Cs. Entering of 137Cs in these bays is connected mainly with atmospheric fall-out and their transportation by the Gulf Stream system [1,2,3,4].

In the Motovsky Bay radioactive contamination is noticeably lower than that observed in the bays entering it. Bottom sediments in the coastal zone of the bay contain 60Co in the range 0.7-1.0 Bq/kg and 137Cs – in the range 5-14 Bq/kg. In the open part of the bay aleurite sediments accumulated still less 137Cs (1-10 Bq/kg) and 60Co (0.5-1.0 Bq/kg). Bases of submarines with nuclear reactor compartments and a repository for the reprocessed nuclear fuel and other radioactive wastes are located in the coastal zone of the Motovsky Bay. In 1997 in the Zapannaya Litsa Bay including Andreev bay 137Cs levels reached 75-115 Bq/kg and those of 60Co – up to 6-33 Bq/kg. It is typical that the whole 20cm thickness of the surface sediments is saturated with 137Cs. The amount of 137Cs and 60Co in the Ara Bay varied in the range from 5 and 13 Bq/kg, correspondingly. 137Cs contents in the water sample from the Zapannaya Litsa Bay was (4.9±0.6) Bq/m³, that in the Ara Bay (4.7±0.4) Bq/m³ [1,4,5,6].

The Kola Bay of the Barents Sea remains the best studied area as for the radioactive contamination [1,2,5,6,7,8].

In 1995-1999, bottom sediments in the Kola Bay almost everywhere contained 60Co at the level of 0.5-1.0 Bq/kg and 137Cs - 1-24 Bq/kg. Bottom sediments activity regularly increased to the atomic bases direction. In the Sajda Bay approximately 2-3 km away from the place of nuclear reactor compartments stay removed from the utilized submarines bottom sediments contained 137Cs and 60Co in the range 3-24 Bq/kg and 0.7-12 Bq/kg, correspondingly. 239,240Pu was also present in the bottom sediments at 1.6 Bq/kg level. In the Olenjya Bay at 800m distance from the ship repairing enterprise “Nerpa” concentration of 137Cs, 60Co and 239,240 Pu is 2-14, 1-2 and 1.6 Bq/kg. The highest level of 137Cs, 152Eu and 154Eu (correspondingly, 60, 55 and 123 Bq/kg) were found in the bottom sediments along the 2 km moorage lines of facilities Sevmorput and Atomflot. Comparatively much 60Co (up to
24 Bq/kg), $^{239,240}$Pu (2.2Bq/kg), $^{137}$Cs (up to 24 Bq/kg) were found in the bottom sediments of the Bays Pala and Ekaterininskaya Gavan in the area of the town of Polyarny. In some samples of bottom sediments an insignificant amount of isotopes $^{134}$Cs, $^{125}$Sb and $^{154}$Eu were also observed. The majority of radioactive substances are localized in the 4-8cm layer of grounds. Evidently, short lived $^{134}$Cs on the Bays’ bottom is of local origin. Appearance of $^{60}$Co and other anthropogenic radionuclides in the sediments samples from the Kola Bay testified to the existence of the directed activity from the working atomic energy installations.

In 1998 in the Kola Bay in the vicinity of the Atomflot base surface waters contained approximately 3-9 Bq.m$^{-3}$ $^{137}$Cs [9].

Algae Lithotamnion corimbossum sampled in the section near Polyarny contained 1.3 Bq/kg $^{60}$Co. Starfishes from the Olenjya contained 0.03Bq/kg $^{137}$Cs. Atlantic cod, haddock, other demersal fish species caught in the Motovsky Bay in the Kildin Island area and at the mouth of the Kola Bay contained extremely little $^{137}$Cs - 0.3-1.0Bq/kg.

Nowadays $^{137}$Cs contents in the bottom sediments of the Kola Bay in comparison to middle 1990-ies is a bit lower and is 10.2±6.7 Bq/kg (Fig). $^{60}$Co up to 4.4 Bq/kg is noticed in some parts of the Northern part of the Kola Bay.

In 2004 $^{137}$Cs contents in the surface waters of the Kola Bay in the vicinity of RTE Atomflot was 4.4.Bq/m$^3$. This index did not change practically in comparison to 1998.

Based on the results of monitoring investigations in 2004 $^{137}$Cs concentration in the common macrophytes species in the Barents Sea bays does not exceed 0.5 Bq/kg dry weight that is cumulative accumulation of caesium by the algae does not take place.

3. Coefficients of caesium distribution for the Kola Bay of the Barents Sea

Together with long- term natural investigations of the radioactive contamination of both: bays and gulfs of the Barents Sea sorption properties of different types of bottom sediments from the Kola Bay in respect to the caesium radionuclides have been carried out in the laboratory experiments. Coefficients values of caesium distribution between water and bottom sediments have been obtained. The greatest values $K_d$ (22.7 in average) are typical of pellite- aleurite silts, for the sediments of the mixed type $K_d$ is lower- 3.4-11.7. The data obtained might add significantly the existing mathematical models of artificial radionuclides behaviour in the marine environment especially at the modelling of distribution of the radioactive contamination during accidents.

4. Classification of bays by the concentration and composition of the artificial sediments in bottom sediments

On the basis of the data on the levels of radioactive contamination in more than in 40 bays a classification of bays by concentrations and composition of artificial radionuclides in the ground has been developed:

The first type includes such bays as Yaryshnaya, Teriberka, Titovka, Ura, Motovsky (Kola Peninsula), Nordensheld, Glazov, Inostrantsev (Novaya Zemlya Archipelago). These bottom sediments have very low $^{137}$Cs content (1-15 Bq/kg) and $^{60}$Co and $^{239,240}$Pu are absent.

The second type includes Dvina, Onega, and Pechora Bays and is characterized by higher $^{137}$Cs concentration; $^{60}$Co is not present in the sediments.

The third type to which the Taganrog, Finish and Bothnia Bays belong is distinguished by a high $^{137}$Cs level (80-400 Bq/kg). Traces of $^{239,240}$Pu (0.2-3.0 Bq/kg), $^{90}$Sr (0.3-9.0Bq/kg) are observed in the grounds but $^{60}$Co is absent.

The fourth type includes Denise and Ob Bays. They receive radioactive discharges from Mayak and other chemical plants. Nuclides concentration is relatively high: $^{137}$Cs-40-100Bq/kg, $^{60}$Co- 0.5-6.0Bq/kg, $^{90}$Sr- 3-16Bq/kg and $^{239,240}$Pu- 1-6Bq/kg.

The fifth type united separate bays (Zapadnaya Litsa, Sajda, Pala) of the Kola and Motovsky Bays area where naval bases are located. The bottom sediments in the deep parts of these bays accumulated 2-9 Bq/kg $^{239,240}$Pu, 30-120 Bq/kg $^{137}$Cs, 4-10 up to 80 Bq/kg $^{60}$Co. The Novaya Zemlya
bays- Abrosimov, Stepovoy and Tsivolka Bays beyond the impact of containers with reprocessed wastes on the ground are also referred to this type. The sixth type includes only one bay- the Chernaya Bay where nuclear tests were carried out. Nowadays contamination of the bay remains strong: $^{137}\text{Cs}$ 50-200 till 1440 Bq/kg, $^{60}\text{Co}$- in the range 92-618 Bq/kg, $^{239,240}\text{Pu}$- approximately 1000-5000 Bq/kg [10, 11].

**FIG. 1:** $^{137}\text{Cs}$ contents in bottom sediments of the Kola Bay of the Barents Sea.

The seventh type includes the areas of the Abrosimov, Stepovoy and Tsivolka Bays where there observed a strong impact on the bottom sediments as nuclear wastes materials are buried in the sediments, besides a local area of the Kola Bay adjacent to the Atomflot is also characterized by extremely high levels: $^{137}\text{Cs}$- 0.6-100 000 Bq/kg, $^{90}\text{Sr}$ 0.30-4000 Bq/kg, $^{60}\text{Co}$ 30-3000 Bq/kg The various images of radionuclides in bottom sediments is partly related to the presence of radioactive (hot) particles [9, 12].

On the whole of analyzing the radioecological situation in the Kola Peninsula bays it should be noted that relatively raised radioactive contamination is observed only in the intermediate vicinity of the atomic bases infrastructure in the bays of the Kola and the Motovsky Bays. The bays lacking on their coasts military bases are practically clean. Considering existence of definite emission of radionuclides from the objects pertaining to the nuclear energy, a state radioecological monitoring of environment and biota must be organized in the Kola and the Motovsky Bays.
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