DEVELOPMENT OF THE JAPANESE TRU WASTE DISPOSAL CONCEPT

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ABSTRACT

This paper summarizes the results of studies performed to demonstrate the long-term, safe geological disposal concept for those TRU wastes produced from a range of fuel cycle activities and having total alpha-radioactivity greater than 1 GBq/ton. The type, amount and characteristics of the wastes are used to develop a suitable disposal facility concept. A generic approach is taken to demonstrate the geological concept, i.e., by incorporating typical Japanese geological conditions and a normal groundwater evolution scenario. The facility concept is based on disposing the waste in rather large tunnels since most of the waste is not exothermic. The waste is categorized into four waste form groups according to their individual characteristics prior to development of the disposal concept. Multi-barrier system combining suitable engineered and natural barriers are considered. The main components of the engineered barrier system are cementitious materials for solidification and filler, and a bentonite buffer. Phenomena that could possibly impact on the disposal system are analyzed by a scenario development process to derive dose calculation cases. Results obtained in the dose calculations show that safe geological disposal of TRU waste with a suitable EBS is possible in a stable deep underground formation with a favorable hydrogeological environment. Future R&D on TRU waste disposal is also discussed and detailed study issues identified with respect to the implementation of TRU waste disposal in the future.

INTRODUCTION

Disposal of wastes containing transuranic (TRU) elements is one of the problems to be solved in the Japanese radioactive waste management programs. These wastes are generated mainly from a range of fuel cycle activities such as reprocessing spent fuel and mixed oxide (MOX) fuel fabrication. TRU-containing wastes have the following characteristics: they contain significant quantities of TRU nuclides with relatively long half-lives; non-TRU nuclides such as I-129, C-14 which are critical to performance assessment analyses; activity levels range from relatively high to low; and the waste occurs in various forms such as cement, organic substances and metals. Studies on the disposal of these wastes have been carried out by several relevant Japanese organizations to establish the disposal concepts and the implementation system including the organization responsible for the disposal.

In the overall waste management program defined by the Atomic Energy Commission of Japan (AEC) [1], the wastes with a total α radionuclide concentration less than a tentative value of 1 GBq/t and relatively low β and γ concentrations are considered suitable for shallow disposal. For wastes with higher α concentration than 1 GBq/t (termed 'TRU waste' in this paper), a specific disposal concept had to be established by the end of the 1990s. In accordance with the AEC program, Japan Nuclear Cycle Development Institute (JNC) and the Federation of Electric Power Companies completed a joint project to integrate the results of their respective research and development activities and to demonstrate the safety of the geological disposal concept for TRU waste. Development of the concept was based to on the Second Progress Report on Research and Development for the Geological Disposal of HLW in Japan (H12 Project) [2] and the particular characteristics of the TRU waste generated in Japan. By the end of 1999 fiscal year, the parties published a joint comprehensive technical review report (TRU report) [3] documenting the results of the project.This report provided the scientific basis for the overall TRU-containing waste disposal program published in March 2000 by the AEC's Advisory Committee on Nuclear Fuel Cycle Backend Policy [4]. The AEC pointed out that TRU-containing waste should be classified into three groups for development of disposal concepts; shallow disposal, mid-depth disposal and deep geological disposal in accordance with their radionuclide concentrations (See Table I).

Table I. Three Candidate Disposal Concepts Considered for Japanese TRU-Containing Waste

Concept	Waste classification
Shallow disposal	Wastes categorized in the same manner as those from operation of nuclear power plants: total α radionuclide concentration less than a tentative value ^{*1} and relatively low β and γ concentrations
Mid-depth disposal	Wastes with high β and γ radionuclide concentration
Deep geological disposal	Others than the above

^{*1} Tentative value; 1 GBq/ton. The value is under consideration.

This paper describes the deep geological disposal concept of TRU waste, summarizing the generation and characteristics of the TRU waste, the geological disposal facility concept and performance of the multi-barrier system.

GENERATION AND CHARACTERISTICS OF THE WASTE

In Japan, most radioactive waste containing TRU nuclides is presently generated at JNC's Tokai Reprocessing Plant and MOX Fuel Fabrication Plant. In the future, radioactive waste will be returned from overseas reprocessing and will be generated from operation of the domestic commercial reprocessing plant and MOX fuel fabrication plant. In the longer-term, decommissioning of reprocessing plants and MOX fuel fabrication plants is anticipated.

TRU-containing wastes occur in various forms such as sludge, ash, metal, organic substances, etc. These wastes were assumed to be treated using current technology for volume reduction, decontamination and solidification and to be solidified with cement and bitumen. The volume of TRU-containing waste generated up to around the year 2035, when the HLW repository is planned to start operation, is estimated to be in the range of 56,000 m^3. It is estimated that of this amount, about 18,000m^3 will have a total α nuclide concentration greater than 1 GBq/t and thus will require isolation from the human environment for a long period. The amount and radioactivity of these wastes are summarized in Table II. The volumes of TRU-containing waste are now being looked at again. It is possible that the volume of waste now categorized for mid-depth geological disposal could be revised if regulatory criteria such as radionuclide concentration limits changed in the future.

Table II. TRU Waste Arisings Assumed for The Study of The Disposal Concept [3]

Classification	Type of waste		Estimated arisings (accumulated) Amount of waste (m^3)
Waste generated from commercial reprocessing	Liquid concentrated I		969
	Ash melt	Spent resin Combustible I Poorly combustible I	31
	Hulls/end-pieces		3638
	Spent iodine filter		238
	Non-combustible I (HEPA, metals)		688
	Non-combustible I (equipment)		1445
Subtotal			7008
Waste generated from JNC reprocessing	Liquid concentrated I (bitumen)		3329
	Liquid concentrated I (after 1998)		2364
	Ash melt	Spent resin Combustible I Poorly combustible I Activated	96
	Hulls/end-pieces		372
	Spent iodine filter		31
	Non-combustible I (metals)		463
	Non-combustible I (others)		463
Subtotal			7,117
Waste generated from operation of the commercial MOX fabrication plant	Combustible		18
	Poorly combustible		12
	Non-combustible		411
Subtotal			442
Waste generated from operation of JNC MOX fabrication plant	Combustible		204
	Poorly combustible		136
	Non-combustible		1655
Subtotal			1995
LLW returned from overseas reprocessing	Compacted solidified waste		902
	Bitumen		270
Subtotal			1172
Waste generated from decommissioning of JNC reprocessing plant	Secondary waste (liquid)		117
Subtotal			117
Waste generated from decommissioning of JNC MOX plant	Secondary waste I (combustible)		6
	Secondary waste II (combustible)		9
	Secondary waste I (poorly combustible)		1
	Secondary waste II (poorly combustible)		2
	Secondary waste I (Non-combustible)		49
Subtotal			68
Total			17,918

DISPOSAL FACILTY CONCEPT

TRU waste has a wide range of properties and therefore has been classified into several groups based on the characteristics of the individual waste types. A suitable design of the engineered barrier system (EBS: waste form, filler and buffer materials) for each waste group was then defined.

Categories of TRU waste

TRU waste were categorized to the following four groups from the view points of the waste characteristics, i.e., the concentration of the radionuclide contained, heat generation, and chemicals which might affect the system performance (Table III).

Table III. Four Groups of TRU Waste

No.	Main waste type	Volume (m^3)	Feature of the group
1	Spent iodine filters	ca. 300	High concentration of I-129, one of the key nuclides
2	Hulls and end-pieces	ca. 4,900	Large inventory of C-14, one of the key nuclides with relatively high heat
3	Liquid concentrated waste, bituminized waste	ca. 7,000	Containing nitrate which will affect water chemistry
4	All the others	ca. 5,800	All the others, low radioactivity concentration

Basic Engineered Barrier System (EBS) components

The disposal concept assumes relatively large underground cavities since most of the TRU waste is not heat producing. The EBS components are expected to have the following functions as a barrier.
Wastes; cementitious materials used to solidify the raw wastes are expected to retard radionuclide release by sorption of radionuclides and due to high-alkaline porewater which decreases the solubilities of some TRUs.
Filling materials; cementitious materials are expected to have the same functions as described above.
Buffer materials; bentonite buffer is expected to have very low permeability, enough to restrict radionuclide migration to diffusion.

For all groups, cementitious materials, which provide the chemical containment function of radionuclides, also serve as fillers in the voids between waste packages and structural materials. Installing bentonite buffer material outside should enhance the physical containment function of the EBS for Groups 1 and 2 with high concentrations of key nuclides like I-129 and C-14. The EBS for Groups 3 and 4 does not require buffer since the main radionuclides included in these groups are TRU isotopes which sorb effectively in both the EBS and the natural barrier system (NBS) and because there is a smaller amount of the key nuclides in these groups.

Underground facilities

Since TRU waste is not significantly exothermic, the disposal concept assumes relatively large underground cavities in order to optimize efficiency and economy. The tunnel type is chosen as the basic design for the disposal cavity, since it has a high construction efficiency and there is a wealth of relevant experience in constructing such large underground cavities in Japan, for example underground power stations, oil reservoirs, etc. The horseshoe-shaped tunnel is chosen for hard crystalline rock, taking into account efficiency for operation and emplacement of waste. On the other hand, the circular cross-section tunnel is chosen for soft sedimentary rock. This shape is more stable than horseshoe-shaped tunnel under the same rock stress conditions, though it is less advantageous operationally with respect to emplacement of the waste packages.

The size and pitch of the disposal tunnels were studied based on mechanical stability and thermal analysis.

Physical rock properties used in the analysis are from the H12 Project [2]. A depth of 500 m was assumed for the facilities in both rock types in this study. In crystalline rock, there is basically no significant constraint on the size in the tunnel stability without concrete lining segment. Relatively small horseshoe-shaped tunnels with square 12 m x 12 m sections and an upper semi-circular space with a 6 m radius are specified for Groups 1 and 2 considering the small waste volume and thermal effect. Large horseshoe-shaped tunnels with square sections 15 m x 20 m and upper semi-circular parts with a radius of 7.5 m are specified for Groups 3 and 4. In sedimentary rock, relatively small circular cross-section tunnels with a radius of 10 m with 0.5 m-thick concrete lining segment are specified for all groups to maintain mechanical stability because sedimentary rock is generally less stable than crystalline rock. In consideration of the performance margin for Groups 1 and 2, buffer material with a thickness of at least 1.2 m is included in both rock cases.

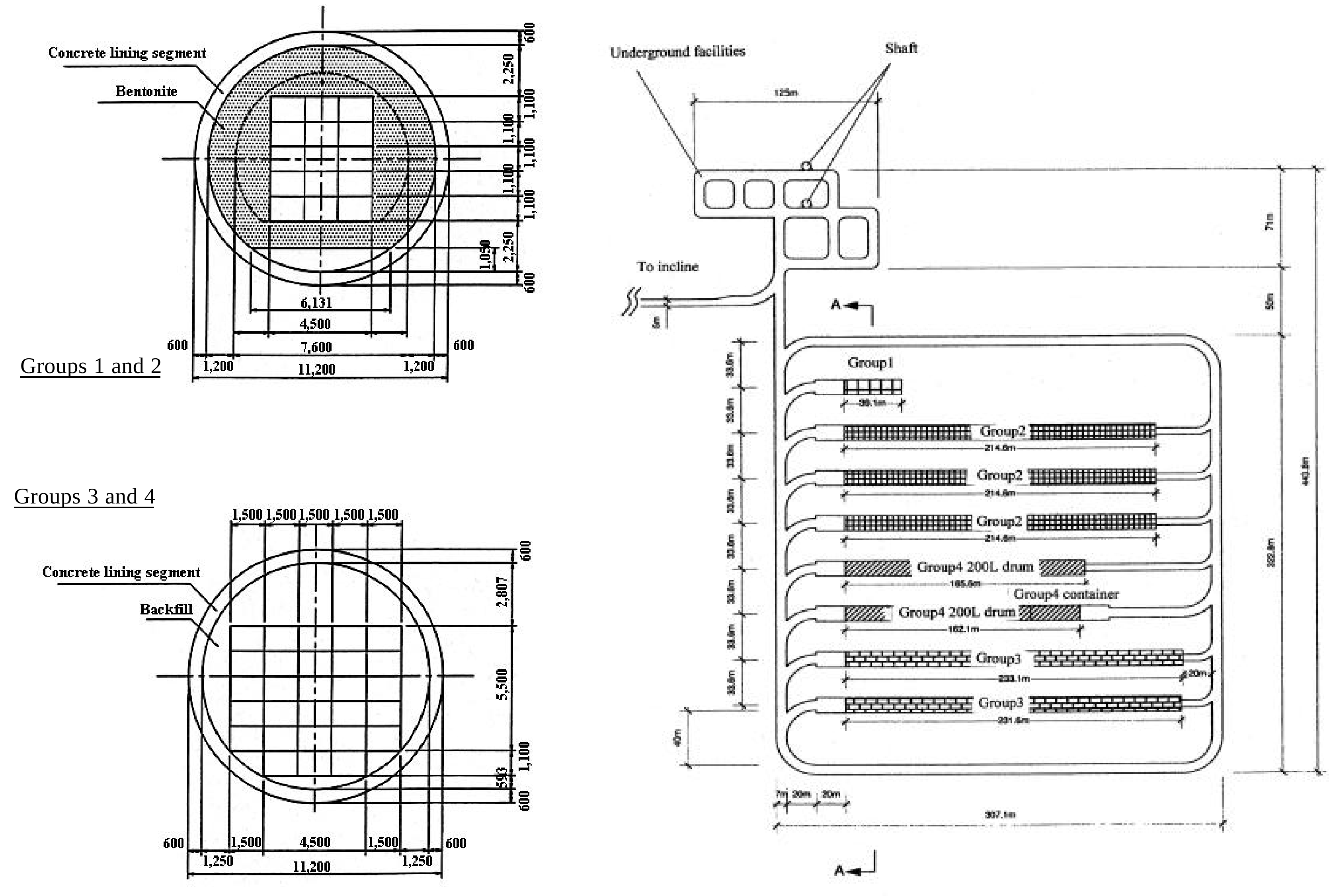
The pitch of the tunnels needed to avoid mechanistic interactions between the tunnels was evaluated using a finite element method (FEM) technique. A pitch 2.5 times the tunnel width and three times the tunnel radius for the crystalline and sedimentary rock cases, respectively was used in the analysis. For the tunnels involving heat-generating Group 2 waste, thermal analysis was carried out to make clear if the above pitches were suitable, because a canister containing hulls and end-pieces was estimated to generate 200 watts when produced. The critical temperature of 80 ℃ for cementitious materials was adopted, at which altered cementitious minerals would be different from those produced at lower temperature [5]. The tunnel pitch necessary for Group 2 in crystalline rock was determined to be 4 times larger than the tunnel width in order to maintain the temperature limit when 25 waste packages (100 canisters) were embedded in the cross-section. On the other hand, the mechanically stable pitch of 3 times the tunnel radius was not affected by the heat analysis for Group 2 tunnels containing 12 waste packages (48 canisters) in the sedimentary rock.

Typical layout of the facilities and cross-sections of the disposal tunnels for the sedimentary rock case are illustrated in Fig. 1.

PERFORMANCE ASSESSMENT

The performance of the disposal system involving the reference geological environment set in the study, the EBS and the disposal facilities discussed previously was evaluated for a normal evolution groundwater scenario. The performance assessment process involves mainly the following phases:

- Establish the geological disposal systems
- Develop scenarios
- Construct models
- Analysis
- Interpret results.



(a) Cross section of the disposal tunnels

(b) Underground facility layout

Fig. 1. Disposal facility concept designed for the sedimentary rock case

The geological disposal system

The geological disposal system examined is a multi-barrier system in which a suitable EBS is constructed in stable, low permeability rock several hundreds meters deep, as described in the former sections. Tunnels with relatively large cross-section are adopted except for tunnels of heat generating Group 2 waste and Group 1 with small inventory. The tunnels for Groups 1 and 2 incorporate bentonite buffer around the cementitious filler to retard the release of the key radionuclides. The tunnels for Group 3, with wastes containing chemicals such as sodium nitrate, could affect the migration of radionuclides released from the other tunnels, are set down gradient in the facilities (See Fig.1.) to avoid the interaction between chemicals in Group 3 waste like sodium nitrate and the radionuclide migration as possible.

Geological environment conditions were selected generically by reference to the H12 Project [2]. The reference case involves the following main parameters:

- Topographic element; Lowlands with hydraulic gradient of 0.01,
- Groundwater chemistry; High pH, low salinity
- Type of rock; Granite with porosity of 2 %, sedimentary Neogen tuffaceous mudstone with porosity of 20 %,
- Hydraulic conductivity; 10^{-9} m/s,
- Geological structural element; Direct flowpath length of 100 m.

Some of the above data were also used in the process of developing the facility design described in the previous section.

Developing scenarios

In order to develop scenarios which should be considered in the performance assessment, important features, events and processes (FEPs) required to be taken into account for TRU waste disposal were identified during a scenario selection procedure. The effect of these FEPs on the disposal environment and nuclide transport were assessed and the uncertainties that may be introduced in the disposal system performance were accounted for by making various assumptions. The Systematic Approach Method [6] developed by SKI (Sweden) was used to involve a comprehensive identification of the various FEPs related to the geological disposal environment, relative importance and potential combinations or mutual influences of the FEPs considered. Then one or more nuclide migration scenarios were developed for use in performance assessment.

The processes in which key environmental conditions and the events that may affect these conditions selected for evaluation via scenario analysis as described above were then evaluated applying computer models, analytical calculations, and doing literature searches, prior to nuclide migration calculations. The following FEPs were evaluated; the thermal and hydrogeological conditions, the radiation field, alteration of cementitious material and bentonite, effects of organics, colloids, microbes, sodium nitrate and gas production on radionuclide transport, and mechanical stability of the system. The results of those evaluations was used to make decisions in handling various analysis preconditions and in selecting important events to be taken into account in the nuclide migration analysis, as well as events of which effects on nuclide migration are not easily determined due to lack of research and data.

In the performance assessment, the reference case groundwater scenario takes into account the evolution of only the EBS with time, assuming no change in characteristics of the natural barrier. The following basic preconditions were assumed:

- The characteristics of natural barrier do not change with time including the distance between the repository and the prevailing groundwater flowpath(s), and the disposal environment remains chemically reducing.
- Volcanic activity, which might increase the disposal facility temperature, is ruled out by site selection.
- A low groundwater flow rate, as might be expected in selected Japanese geological repository environments, is assumed.

In addition, the following items were selected and considered in the scenario development as important events that could occur in the EBS, based on the evaluations of the FEPs described above:

- Changes in near-field chemical and hydraulic environments due to cement alteration
- Alteration to calcium-type bentonite
- Expulsion of cement porewater due to gas generation
- Effects of nitrate on nuclide sorption and elemental solubility
- Alteration of bentonite and rock due to highly alkaline cement leachate

Combining those effects with the groundwater flow, radionuclide transport and established disposal system features gives the following groundwater scenario.

- Groundwater enters the disposal facility
- Radionuclides and other components of the cement waste forms such as sodium nitrate, as well as cementitious minerals themselves, are dissolved in the groundwater at rates determined by dissolution rates of solid phases or elemental solubility limits
- Dissolved radionuclides migrate through the EBS by advection and/or diffusion with retardation due to sorption onto solid phases en route to the geosphere.
- Radionuclides migrate through the geosphere in advecting groundwater with retardation due to sorption or matrix diffusion en route to the biosphere.
- Radionuclide concentrations in the groundwater may be reduced by sorption, precipitation, dispersion, dilution and radioactive decay before the radionuclides reach accessible waters and pass into the food chain, thus resulting in a dose to man.

The following five perturbing factors change the functions of the barriers:

- The dissolution of the cement components alters the mineralogy and the porewater composition, affecting the solubility and sorption of nuclides.
- Metal corrosion produces hydrogen gas, which expels the porewater in the EBS.
- Through reacting with cement components, bentonite calcification (conversion to a calcium-form) occurs.
- High concentrations of sodium nitrate reduce the sorption of anionic nuclides like I-129 and may affect elemental solubility.
- Highly alkaline water influences the sorption of nuclides onto rocks due to the alteration of rocks to form phases like, for example C-S-H, which have different sorption properties.

Analysis

The nuclide transport analysis cases are established such that they consider the uncertainty in the effect of each event included in the reference groundwater scenario mentioned above. In order to effectively combine the five perturbing factors, the following measures were taken to establish 13 analysis cases.

- As a reference base case, the effects of gas generation, cement alteration and bentonite calcification were assumed considering time-independent conservative parameters.
- Cases where time-dependent parameters were used for the above three effects were also considered.
- Cases that do not involve alteration of the EBS were established for direct comparison with the cases that result in alteration.
- Cases that involve each of the effects of gas generation, cement alteration and bentonite calcification were considered separately to clarify the impact due to each effect.
- The effect of alteration of host rocks is considered by assuming the formation of C-S-H only.
- The effect of nitrate was considered in a very simple manner as a first scoping study.
- The failure of the buffer is assessed in a sensitivity analysis.
- Distribution coefficients for I-129 and C-14 for sorption onto rock are varied in a sensitivity analysis.

The migration of nuclides through the EBS and the geosphere was then calculated using a computer code which solved one-dimensional advection-dispersion equations with a finite differential method for the EBS region and an analytical solution for the natural barrier region, based on the conceptual model of the groundwater scenario. A river water scenario was adopted as the biosphere model and the safety of the disposal system was expressed in terms of effective dose. Details of the model, equations and input parameters used in the assessment are found in TRU report [3].

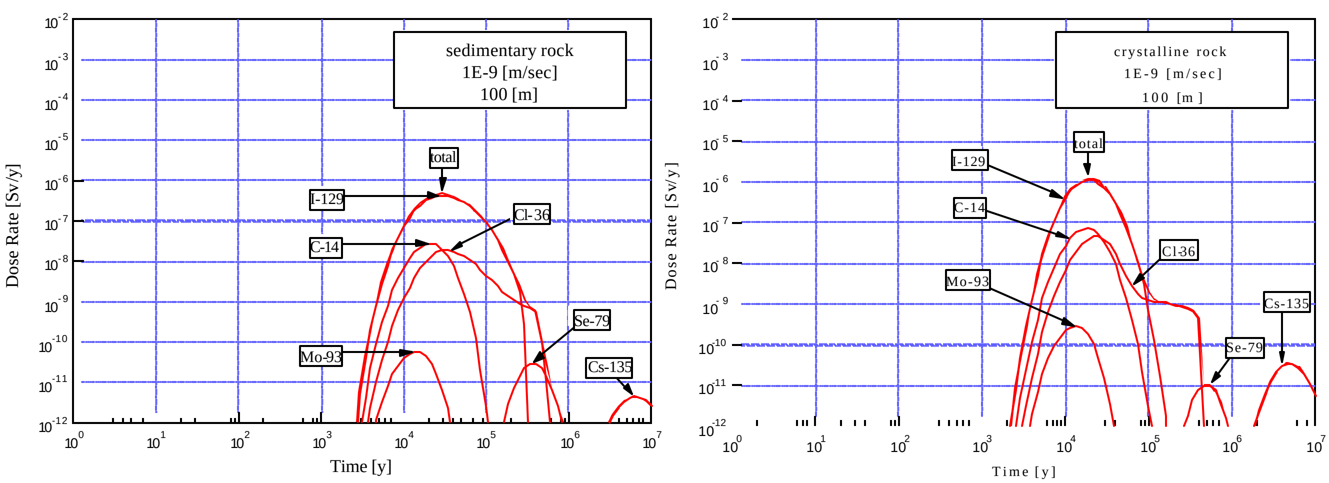
Interpretation of results

Dose for the base case (Case 0), where the effect of gas generation, cement alteration and bentonite calcification are conservatively considered, was evaluated for a 100 m migration path length in the natural barrier with a hydraulic conductivity of 10^{-9} m/s and a gradient of 0.01, as a reference case.

The analysis results for Case 0 are shown in Fig. 2(a) and 2(b) for sedimentary and crystalline rocks, respectively. Maximum values for dose in each waste group and the total dose are shown in Fig. 3(a), (b) for a part of the analysis cases. In addition, the range of safety guidelines for geological disposal in other countries is also shown.

In most of the cases, the waste form group contribution for both rock types was as follows: Group 1>Group 2>Group 3>Group 4.

In all waste groups, the dominant nuclides were either C-14 or anions that have small distribution coefficients (Kd) in both the engineered and natural barriers. Iodine-129 of Group 1, contained in spent iodine filters to fix I-129 in the form AgI, does not decay sufficiently because of its long half-life of 17 million years, and thus is the most significant radionuclide. Maximum dose rate due to I-129, however, shows little change even when no sorption is assumed since this radionuclide is not well sorbed on rocks. Other non-sorbing radionuclides with shorter half-lives like Cl-36 may have a similar migration tendency to I-129. A higher performance waste form for immobilizing I-129 would retard the radionuclide in the EBS enough to decrease the dose (See the latter section). The contribution of the second most significant nuclide, C-14, to the total dose is one order of magnitude less than I-129. Chemical form of C-14 released from activated "hull" and "endpiece" wastes of Group 2 is important in setting realistic values of Kd and solubility of carbon in the performance analysis because pessimistic assumptions of organic form [7] leads to very small Kd. The performance of the natural barrier, i.e., Kd of the rock, permeability, and path length are sensitive to the contribution of C-14 to total dose due to the additional effect of decay (half-life 5,730 years) during the migration. One example is shown in the case in which the path length is compared in Case 1 (500 m) to Case 0 (100 m). Actinides do not show a significant contribution to dose because they have high sorption coefficients and their concentrations in the groundwater through the disposal system are very low for all the groups.



(a) sedimentary rock (b) crystalline rock

Fig. 2. Calculated dose rate in the base case (Case 0)

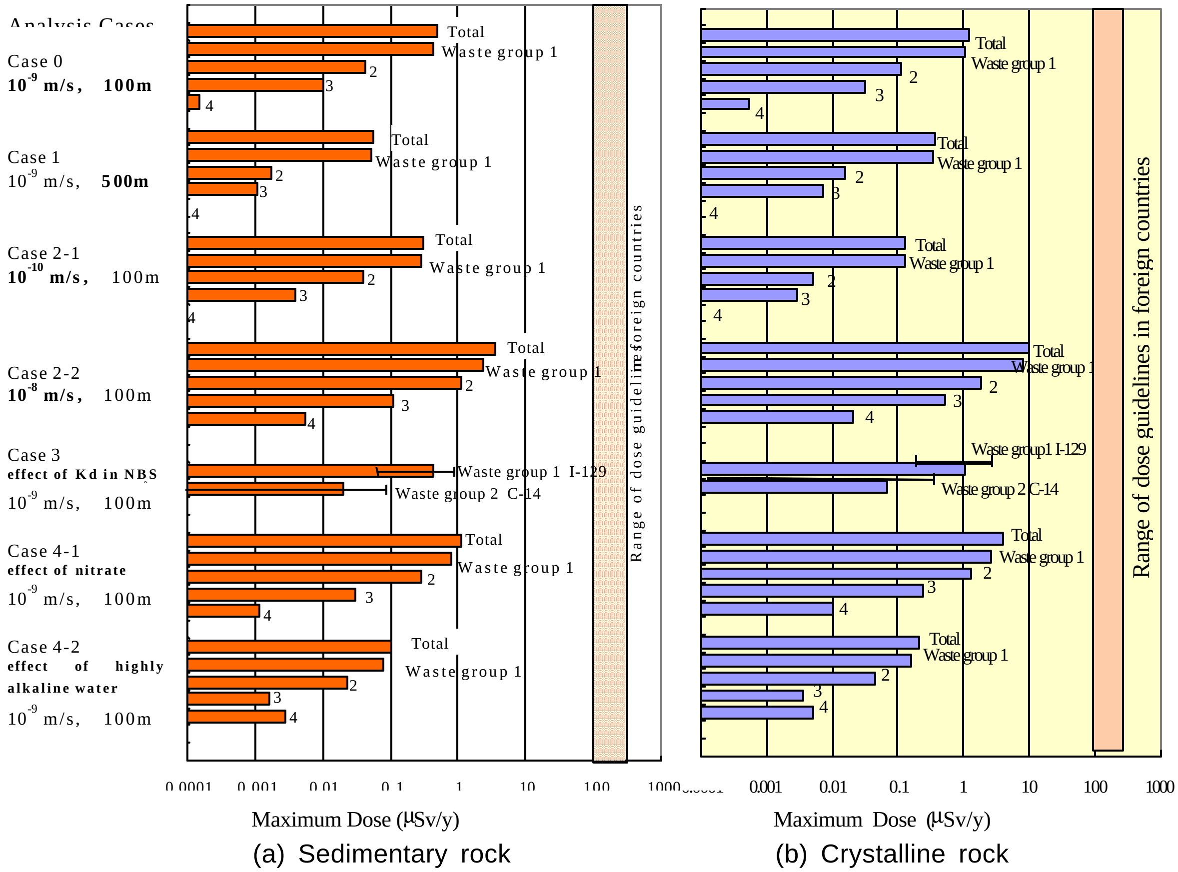


Fig. 3. Maximum dose of each analysis case

The dose shows significant dependence on the hydraulic conductivity (i.e. the groundwater flow velocity). Total dose is about 10 μ Sv/y with a hydraulic conductivity of 10^{-8} m/s in the crystalline rock case (Cases 2-1, 2-2). Sedimentary rock is less sensitive to hydraulic conductivity below 10^{-9} m/s because transport is controlled by diffusion due to its large porosity, 20 %, under this geological condition.

Considering the effect of sodium nitrate generally increases the dose, since very small Kd values of the key nuclides on cementitious material, bentonite and rocks were assumed. For the natural barrier, no sorption of I-129 and C-14 is considered but this has little effect on dose (Case 4-1). A similar tendency is shown in the sensitivity analysis of rock Kds for I-129 and C-14 (Case 3). Reducing sorption to zero showed little effect on calculated doses since the Kds for these nuclides sorption onto the rocks were set to very small values, 0.001 m^3/kg, in the base case.

On the other hand, an optimistic assumption of C-S-H formation along the pathway in the rock gives an effective reduction of the total dose, since the increase by one order of Kds of I-129 and C-14 on rocks is assumed, i.e. equal to the Kd for the cementitious material of which main mineral is C-S-H (Case 4-2).

The failure of the function of the buffer was examined as a very pessimistic case of bentonite alteration. The result showed no affect on the dose because most of I-129 had been released from the EBS by the time the maximum dose appeared. This means that the geological conditions of the base case are rather good when comparing the retardation function of the bentonite buffer for I-129 migration. The function and effectiveness of the buffer should be discussed for a wide range of the geological conditions in the future.

In the Japanese disposal concept proposed [3,4], maximum dose resulted for the case with a hydraulic conductivity

of 10^{-8} m/s for crystalline rock, with a value of about 10 μSv/y, which is sufficiently below the safety regulations in other countries. For this study, the analysis was limited to the reference case of the groundwater scenario with considering only the EBS evolution. Alternative cases based on the H12 Project [2], which cover changes in geological conditions, uplift and climatic change, are not considered in the present study but are identified as a topic for future investigation.
The performance analysis studies of the effects of the above characteristics on nuclide migration in the multi-barrier system demonstrated the long-term safe of geological disposal of TRU waste; a suitable engineered barrier system is feasible in a stable, deep underground formation under favorable hydrogeological conditions. More detailed analysis considering a broader scenario range is required in the future.

FUTURE ISSUES

In the former section, it was shown that long-term, safe geological disposal of TRU waste in a suitable EBS is possible in a stable deep underground formation with an appropriate geological environment. Nevertheless, more detailed analyses and assessments are required. The regulatory limits of radionuclide concentration for each disposal concept will be discussed by the Nuclear Safety Commission of Japan (NSC). Further investigation of matters such as the disposal implementation system, costs, security and the establishment of safety regulations is required to implement disposal. It is also important to carry out disposal in a phased manner, gaining the understanding of society and building confidence in the disposal concept by an open program of research, development and demonstration.

Future research and development on TRU waste disposal is also discussed by the AEC's Advisory Committee and detailed study issues have been categorized according to the following [3]:
1) Enhance the reliability of safety assessment,
2) Further rationalization and detailed examination of the disposal technology concept,
3) Study of more advanced waste treatment technology, and
4) Enhancement of waste database.

The above issues are broken down in this paper as in the following.

Topics for future research and development

Further rationalization and detailed examination of the disposal technology is needed to establish detailed specifications for disposal facilities needed in order to estimate costs. Further enhancement of the reliability of safety assessment should take into account progress in areas such as waste categorization and establishment of safety regulations.

More advanced waste treatment technology, aimed at improving the solidified waste properties for a relevant range of geological environments in Japan is also desirable. Furthermore, enhancement of waste databases and improvement of analytical methods for measuring radionuclide concentrations are essential and a major prerequisite for safety assessment. More specific requirements include (see Table IV):

(1) Enhancement of the reliability of safety assessment
The analysis and assessment performed in this report have revealed that soluble, poorly sorbed nuclides such as I-129 and C-14 dominate disposal system performance. It is important to obtain more reliable nuclide migration data for relevant disposal environments. Conservative models and data were used for the assessment of specific phenomena such as long-term dissolution of cement, development of a high-pH plume, degradation of bentonite in a high-pH environment, gas generation and migration, the effect of nitrate, etc. These items require examination with more realistic models and data. Improvements will require supporting laboratory studies, possibly confirmed (validated) by field tests and natural analogue studies.

(2) Further rationalization and detailed examination of disposal technology
The present study showed that concentrated disposal in a large underground cavern is possible for TRU waste, because of its low heat generation. The design and construction of disposal facilities are optimized for this type of

concept by classifying waste into different groups according to waste properties and constructing engineered barriers appropriate to these groups. Further rationalization and optimization of disposal technology is needed to clarify detailed disposal facility specifications. Such specifications will allow better definition of disposal costs, identification of requirements for waste categorization and establishment of safety regulations. At the same time, investigations are required to clarify the performance and degradation of cementitious materials (used for waste solidification and as filling material) and bentonite buffer materials. Depending on the results of a detailed assessment of alkaline degradation of bentonite, development and application of low-pH concrete may be required or alternative disposal concepts/strategies investigated. The optimization of disposal technology will be associated with iterative performance assessment in the future.

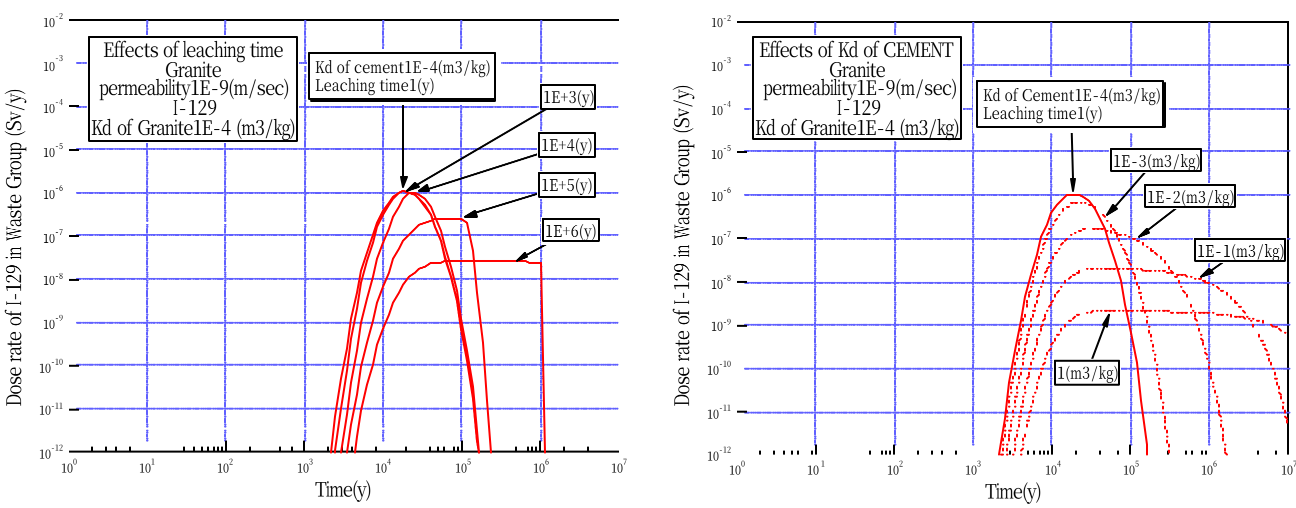
(3) More advanced waste treatment technology

With regard to filters containing large quantities of I-129, safe disposal is considered to be possible with current cement solidification technology, but the performance depends on the assumed geological environment. A more advanced waste treatment technology would improve flexibility in providing acceptable performance in a wider range of geological environments in Japan. Fundamental research into iodine solidification technologies such as AgI glass solidification, HIP sintered silica-gel, high-performance cement, HIP with copper powder and sodalite are currently being carried out and some of basic results have been reported in a national conference. Figures 4(a) and 4(b) show the results of sensitivity analysis of dose dependence on the leaching period of solidified waste and the Kd value of solidification material. If the leaching period of solidified waste can be extended to approximately one million years, or if the Kd value of solidification material improves by a factor of one thousand, then the resultant dose is calculated to be reduced by a factor of approximately one hundred.

(4) Enhancement of waste databases, etc.

An inventory of nuclide concentrations and chemical forms in waste is a major prerequisite for disposal safety assessment. Enhancement of database quality for all types of waste is required. Highly reliable examination methods, including non-destructive measurement technology, need to be developed.

The relevant organizations shall continue to work closely while separating duties appropriately and efficiently to solve the above problems. It is important to integrate and summarize research results appropriately to be reflected in the establishment of an implemented disposal system and provision of safety regulations, which are now being formulated by the NSC's Advisory Committee.



(a) Dose dependence on the leaching period of solidified waste

(b) Dose dependence on Kd value

Fig. 4. Results of sensitivity analysis assuming high performance immobilization of AgI.

Table IV. Topics for future research and development

Item	Study issue
1.Enhancement of reliability of safety assessment	
1.1 Improvement of models/data on specific phenomena	
Migration data on soluble, poorly sorbed nuclides such as .^{129}I, ^{14}C, etc.	·Reliable data acquisition on Kd, D of ^{129}I, ^{14}C,etc. ·Identify chemical forms of ^{14}C in relevant systems ·Improvement of thermodynamic database of cement environment
Degradation of cement and effect of highly alkaline water	·Effect of carbonation and nitrate on cement degradation. ·Reliable data acquisition on Kd, D, etc. of altered cement ·Effect of high pH plume on rock
Degradation of bentonite	·Reliable data on Kd, D etc of altered bentonite ·Zeolitization, CSH formation rates ·Examine practicality of low-pH cement
Waste leaching properties	·Confirm nuclide release mo del (congruent to metal corrosion). ·Data acquisition on long-term corrosion rate of waste metal.
Gas generation and migration	·Examine gas generation from metal corrosion ·Validate model/data on microbial effects on organics. ·Development of discrete model on gas migration
Effect of organics, nitrate and microbes	·Identify degradation products of asphalt ·Data acquisition on effect of nitrate on nuclide migration ·Improvement of data on microbial effect
1.2 Integrated performance assessment	
Scenario analysis	· Examine perturbation scenarios(refer to H12 Project)
Environment analysis	· Application of improved models/data
Nuclide migration analysis through EBS	· Examine the effect of cracking in concrete structures on nuclide migration
Nuclide migration analysis through NBS	·Examine heterogeneous porous media ·Application of fracture network model (refer to H12 Project)
Biosphere assessment	·Examine other scenarios (refer to H12 Project)
2. Further rationalization and detailed examination of disposal technology	
2.1 EBS technology	
Mechanical properties of EBS Materials	·Improvement of database on mechanical properties of altered cement and bentonite · Development of low-pH cement.
Structural analysis of EBS	·Validate models/data for structural analysis. ·Data acquisition of long-term creep of rock and long-term mechanical properties of altered EBS materials ·Examine effects of cement degradation and nitrate on EBS structure
Construction technology for EBS	·Alternative specification of buffer ·Demonstration of construction technology of EBS
2.2 Repository design and construction technology	
Specifications	·Re-estimation of waste volume
Design analysis of disposal facility	·Detailed design of repository by considering heterogeneity of rock, etc.
Construction technology	· Detailed examination on specification of disposal tunnels and construction technology (alternative tunnel type including silo type).
Operation system	·Detailed examination of waste transportation and handling systems ·Development of waste package systems
Backfill technology	·Detailed specification of backfilling materials and technologies.

Table IV. Topics for future research and development (Continued)

Item	Study issue
3. More advanced waste treatment technology	
Solidification technology, volume reduction technology	•High performance solidification technology for iodine (AgI glass, HIP with copper powder, high performance cement, sodalite, etc.) •Melting technology, etc.
4. Enhancement of waste database, etc.	
Enhancement of waste database quality	•Enhancement of database on nuclide concentration and chemical form, etc.
Establish highly reliable examination method	•High performance non-destructive measurement technology. •Development of estimation method for nuclide inventories which are difficult to measure directly.

SUMMARY

As a result of performance assessment studies, taking into account the specific effects of TRU waste on nuclide migration in the multi-barrier system, it has been shown that long-term, safe geological disposal of TRU waste with a suitable engineered barrier system is possible in a stable deep underground formation with a favorable hydrogeological environment. However, more detailed analyses and assessments considering a broader scenario range are still required.

According to the overall TRU waste disposal program published in March 2000 by the AEC's Advisory Committee on Nuclear Fuel Cycle Backend Policy [4], further consideration of matters such as the disposal implementation system, costs, funding and the establishment of safety regulations is required before implementing disposal. With regard to the research and development, all relevant organizations are required to continue studies, using the staff and facilities of individual organizations by separating duties appropriately and efficiently. It is also important to carry out the implementation of disposal in a stepwise manner, in order to gain the understanding of society and to build confidence in the disposal concept by a structured program of assessment and investigation.

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