Chapter 5 Outline of the disposal system and safety assessment for concrete vault disposal and intermediate-depth disposal

5.1 Outline of facilities for concrete vault disposal and intermediate-depth disposal

In this section, concrete vault disposal for low-level waste and an intermediate-depth disposal system (e.g. 50−100 m from the surface) documented in an AEC report on backend policy (AEC, 1998) are summarised briefly.

(1) Summary of preconditions

a. Disposal outline

The disposal concept of radioactive waste in Japan is divided into management-type and isolation-type (NSC, 1985).

Shallow disposal belongs to the management type and has already been implemented as concrete vault disposal for solidified radioactive waste from nuclear reactors. For waste which has higher concentrations of radioactive substances, disposal at intermediate depth is planned as one of the disposal options.

Land disposal of low-level waste has been implemented following the basic concept that radioactive waste can be contained by engineered structures such as a concrete vaults at shallow depths. The concrete vault disposal design assumed for low-level waste with nuclide concentrations below an upper limit specified by the government is shown in Figure 5.1-1.

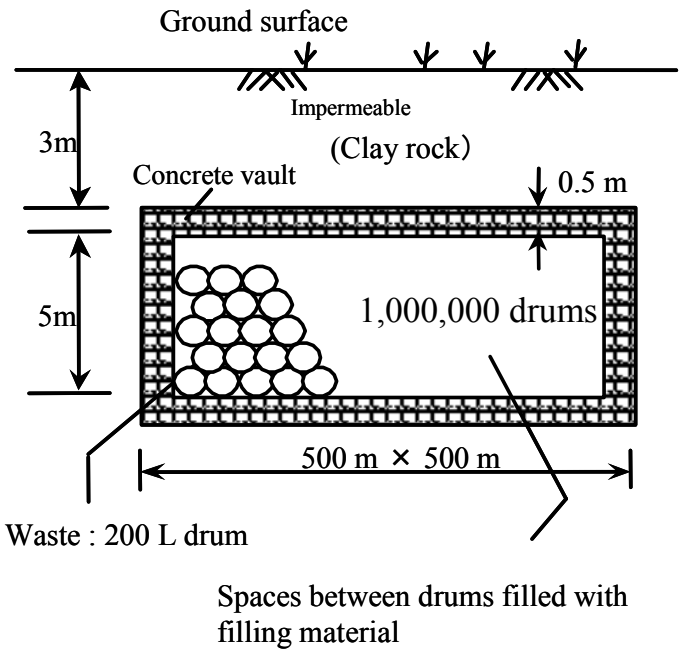

Figure 5.1-1 Outline of concrete vault disposal

Additionally, an outline concept of intermediate-depth disposal in Japan is shown in Figure 5.1-2.

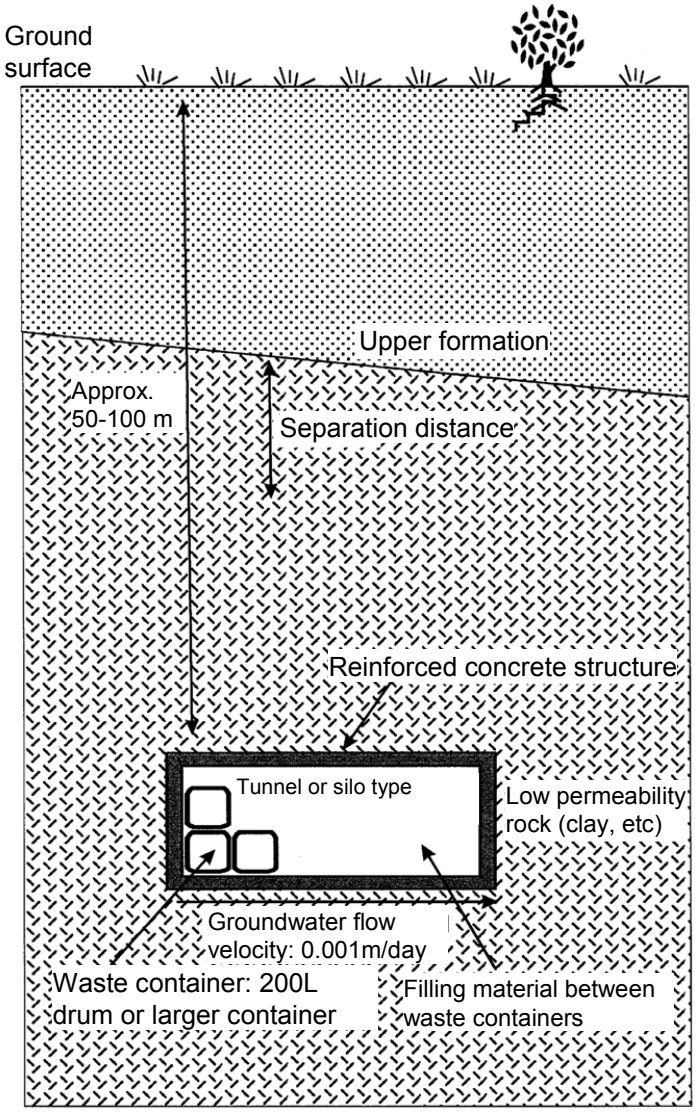

Figure 5.1-2 Outline of intermediate-depth disposal

(i) The waste is disposed of at a depth at which underground facilities can be constructed (e.g. about 50 – 100 m from surface), taking underground conditions into consideration.

(ii) Rocks with the ability to prevent radionuclide transport are selected.

(iii) A disposal facility which has the ability to contain radionuclides, such as a concrete vault, is constructed.

(iv) Considering the decrease in radionuclide concentrations due to decay, the disposal facility is managed for a period of several 100 years.

b. Outline of the disposal system

In a concrete vault disposal facility, multiple barriers are specified. These combine clay and cementitious barriers, which are constructed taking into account the features of the site. A multiple barrier system

covered with a thick layer of soil has been evaluated. A diagram showing an outline of concrete vault disposal at the Rokkasho Low-Level Radioactive Waste Disposal Center is shown in Figure 5.1-3.

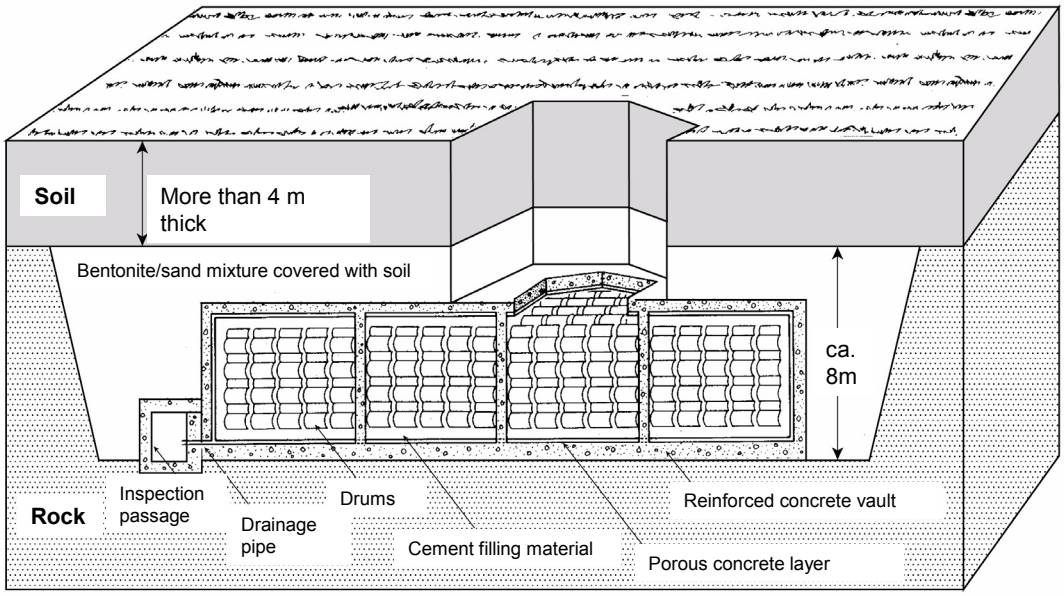

Figure 5.1-3 Outline of concrete vault disposal (NSC (ed.), 2000b)

In the Japanese concept, the concrete structure is constructed underground at a depth of several tens of meters. The structure is tunnel-type or silo-type and is based on the SFR in Sweden and VLJ in Finland. It is assumed that solidified waste is stored in disposal packages and that the facility is backfilled.

Radionuclide migration in groundwater after the management period is evaluated in the same report. It is assumed that radionuclide migration can be retarded in natural soil and through improvement and combination of impermeable materials.

(i) Restriction of migration by natural soil, etc.

- Locating the disposal facility underground in a geological formation with low permeability and a small hydraulic gradient will decrease the flux of groundwater and the transport velocities of radionuclides. Placing the disposal facility at a sufficient distance from the surface will ensure that travel times are sufficiently long for nuclide concentrations to decrease in the case where radionuclides are released from the disposal facility.

(ii) Restriction of migration by impermeable material

- The release of radionuclides from the disposal facility depends on the flux of groundwater and the diffusion rate in the facility. Hence, by ensuring that the environment of the facility contains

impermeable materials such as bentonite clay, the flow of groundwater in the facility is decreased and the diffusion of radionuclides is prevented.

(2) Structure of the engineered barriers

For concrete vault disposal, general geological environment conditions are specified in this report and specific upper limits for nuclide concentrations are derived from current government regulations. Considering these constraints, the following components are considered:

- Waste (waste + cement mortar + steel package)
- Filling material (cement mortar)
- Concrete vault (reinforced concrete)

For disposal at intermediate depth, the disposal facility have the same or a greater capability (than shallow disposal) to contain radionuclides in an AEC report and it is assumed that the disposal facility will be constructed at a sufficient depth, based on the No. 1 and No. 2 underground facilities at the Rokkasho Low-Level Radioactive Waste Disposal Center.

Considering the above, the intermediate-depth disposal facility consists of the following components.

- Waste (waste + cement mortar + steel package)
- Filling material (cement mortar)
- Concrete vault (reinforced concrete)
- Bentonite clay

5.2 Future plans for safety assessment of concrete vault disposal and intermediate-depth disposal

For low-level solid waste generated from nuclear reactors, the basic safety assessment methodologies for concrete vault disposal and intermediate-depth disposal have already been published as follows: policy of radioactive waste processing (AEC, 1985); backend policy (AEC, 1998); reference values for safety regulation of low-level solid waste land disposal (NSC, 1986); reference values for safety regulation of low-level solid waste land disposal (NSC, 1992).

In the safety assessments for concrete vault disposal and intermediate-depth disposal in this report, the safety of TRU waste disposal for each category described in Chapter 2 is evaluated. This report is aimed only at evaluations following the operation period. Additionally, it is outside the scope of this report to evaluate phenomena that are not considered to be likely, for example the drilling scenario for intermediate-depth disposal.

5.3 Scenario evaluation for concrete vault disposal and intermediate-depth disposal

5.3.1 Assessment scenarios and models for concrete vault disposal

The following scenarios were considered here:

- Construction scenario
- Residential scenario
- Groundwater migration scenario

The details of assessment scenarios and models are shown in Appendix 5A.

5.3.2 Assessment parameters for concrete vault disposal

The amounts of radioactive substances in the waste are based on values given in Table 2.4.2-2. The other parameters that are necessary for the evaluation use values specified in NSC reports. However, a dose conversion factor from ICRP Pub.72 (IAEA, 1995) is used. Since there are no values specified for daughter nuclides, these were specified with reference to NSC reports, as shown in Table 5.3.2-1) (e.g. IAEA, 1994).

In the groundwater migration scenario, the release coefficients of C-14, Cl-36 and I-129 and sorption distribution coefficients for transport pathways are specified by considering the data and parameter ranges for oxidizing groundwater in geological repository assessments. These values are shown in Table 5.3.2-2.

Table 5.3.2-1 Newly specified assessment parameters

Element	Enrichment coefficient for crop ((Bq/kg)／(Bq/kg))				Transport coefficient for animal products (d/kg)					Enrichment coefficient (-)	Release coefficient (-)	Sorption distribution coefficient (m^3/kg)	
	Crop	Rice	Leafy vegetables, non-leafy vegetables, fruit	Fodder	Eggs	Milk	Beef	Pork	Chicken	Seafood	Repository	Transport pathway	Agricultural soil
Pb	3.4E-02	4.0E-03	3.4E-02	1.1E-03	1.2E+00	3.0E-04	4.0E-04	3.1E-02	1.2E+00	3.0E+02	3.0E-04	1.0E-01	2.2E+01
Po	1.5E-03	2.0E-03	1.5E-03	9.0E-02	9.9E-04	3.4E-04	5.0E-03	9.9E-04	9.9E-04	5.0E+01	3.0E-04	1.0E-01	6.6E+00
Ra	8.0E-03	6.6E-04	8.0E-03	8.0E-02	2.0E-05	1.3E-03	9.0E-04	3.5E-02	4.8E-01	5.0E+01	3.0E-02	5.0E-02	2.4E+00
Ac	1.0E-03	1.0E-03	1.0E-03	4.0E-03	2.0E-03	2.0E-05	2.0E-05	1.0E-02	4.0E-03	2.5E+01	3.0E-04	1.0E+00	5.4E+00
Th	3.5E-03	1.9E-05	3.5E-03	1.1E-02	2.0E-03	5.0E-06	1.0E-04	1.0E-02	4.0E-03	1.0E+02	3.0E-04	1.0E+00	8.9E+01
Pa	4.0E-02	4.0E-02	4.0E-02	1.0E-01	2.0E-03	5.0E-06	1.0E-03	1.0E-02	4.0E-03	1.0E+01	3.0E-04	1.0E+00	6.6E+00

Note 1:Reference

IAEA TRS No.364 (IAEA, 1994), IAEA S.S No.57 (IAEA, 1985), PNL-3209 (Napier et al., 1980)

Table 5.3.2-2 Revised assessment parameters

Element	Release coefficient(-)		Sorption distribution coefficient of the transport pathway (m^3/kg)	
	Base case *1	Variation case	Base case*1	Variation case
C	0.1	1.3*2	0.01	0.0001*2
Cl	0.1	2.9	0.01	0.0001
I	0.1	0.23	0.01	0.0001

*1: Values specified NSC reports are used.

*2: Specified values for organic carbon.

5.3.3 Assessment scenarios and models for intermediate-depth disposal

A groundwater migration scenario is considered to provide a plausible pathway from the disposal facility by which a radiological dose may be delivered to the public.

The dose pathway in the groundwater migration scenario considers the following 4 routes:

- Channel for river water use
- River bank construction route
- River bank residential route
- River bank agriculture route

Details of the assessment scenarios and models are shown in Appendix 5C.

5.3.4 Assessment parameters for intermediate-depth disposal

The amounts of radioactive substances in the waste are given in Table 2.4.2-2. Dose conversion factors and data for the daughter nuclides of α-emitting nuclides are the same as for concrete vault disposal described in Section 5.3.2. As in the variation case, the leaching rate of metal and amounts of water that infiltrate the repository were revised and are shown in Table 5.3.4-1.

The target waste includes activated metal. However, the flux of inflowing groundwater is expected to be reduced by the emplacement of bentonite clay in the intermediate-depth disposal facility. Therefore, a decreased metal leaching rate and a decreased amount of groundwater infiltration are considered.

The leaching rate of metal waste is estimated from the corrosion rate and surface area of each material (RWMC, 2002). They are specified for CB (the main material is zircaloy) and BP (the main material is stainless steel). The amount of groundwater infiltration is estimated from the coefficient of hydraulic conductivity of the surrounding host rock (1.0×10^{-7}m s^{-1}) and by assuming that the facility has the same characteristics as the surrounding geological environment in the 3rd interim report. Hence, it is assumed

that the transmissivity of the facility is decreased by 2 orders of magnitude by the emplacement of bentonite clay, which has a permeability 2 orders of magnitude lower.

Table 5.3.4-1 Revised assessment parameters

Parameter		Base case	Variation Case
Leach rate (l/y)	CB	Instant release	1.0×10^{-5}
	BP	Instant release	3.0×10^{-4}
Transmissivity for water inflow to the facility ($m^3/m^2/y$)		7.3×10^{-2}	7.3×10^{-4}

5.4 Safety assessments of concrete vault disposal and intermediate-depth disposal

5.4.1 Safety assessment of concrete vault disposal

Based on the assessment scenario and parameters summarized in Section 5.3, a dose assessment was carried out for the products of α-particle decay. The results are summarised in Table 5.4.1-1 and Figure 5.4.1-1.

Of the three assessment scenarios, the residential scenario shows the highest value and the dose after 300 years is estimated to be 8.4 μSv/y. The dominant nuclides are the spent fuel fission products Sr-90 and Am-241 that are present in the waste.

The construction scenario and groundwater migration scenario gave lower doses than those of the residential scenario. Values of 0.025 μSv/y and 0.01 μSv/y are estimated. In the construction scenario, the dominant nuclides are Cs-137, Sn-126 and Sr-90. In the groundwater migration scenario, C-14 was the dominant nuclide.

The dose in the variation case of the groundwater migration scenario was significantly affected by a decrease in barrier performance with respect to C-14. The dose was about 2 orders of magnitude larger than that of the base case and was estimated to be 6.9 μSv/y. However, the evaluation did not consider the decrease in transmissivity of the facility caused by using bentonite clay. Taking this factor into account, it is possible that the calculated dose is larger than the actual likely dose.

TRU waste contains many nuclides that significantly affect internal radiation exposure and are not easily transported by groundwater. Therefore, it is necessary for the residential scenario in the human proximity scenario to focus on concrete vault disposal. In the residential scenario, it is not expected that there will be a strong barrier function that decreases the dose. Therefore, it is important that the concentrations of radioactive substances in the emplaced waste are adequately evaluated and managed.

Table 5.4.1-1 Results of dose assessments for concrete vault disposal

Assessment scenario		Dose (μSv/y)	Time (y)	Dominant nuclide
Residential scenario		8.4×10^{0}	3.0×10^{2}	Sr-90, Am-241, Tc-99
Construction scenario		2.5×10^{-2}	3.0×10^{2}	Cs-137, Sn-126, Sr-90
Groundwater migration scenario	Base case	1.0×10^{-2}	1.6×10^{4}	C-14
	Variation case	6.9×10^{0}	5.4×10^{2}	C-14

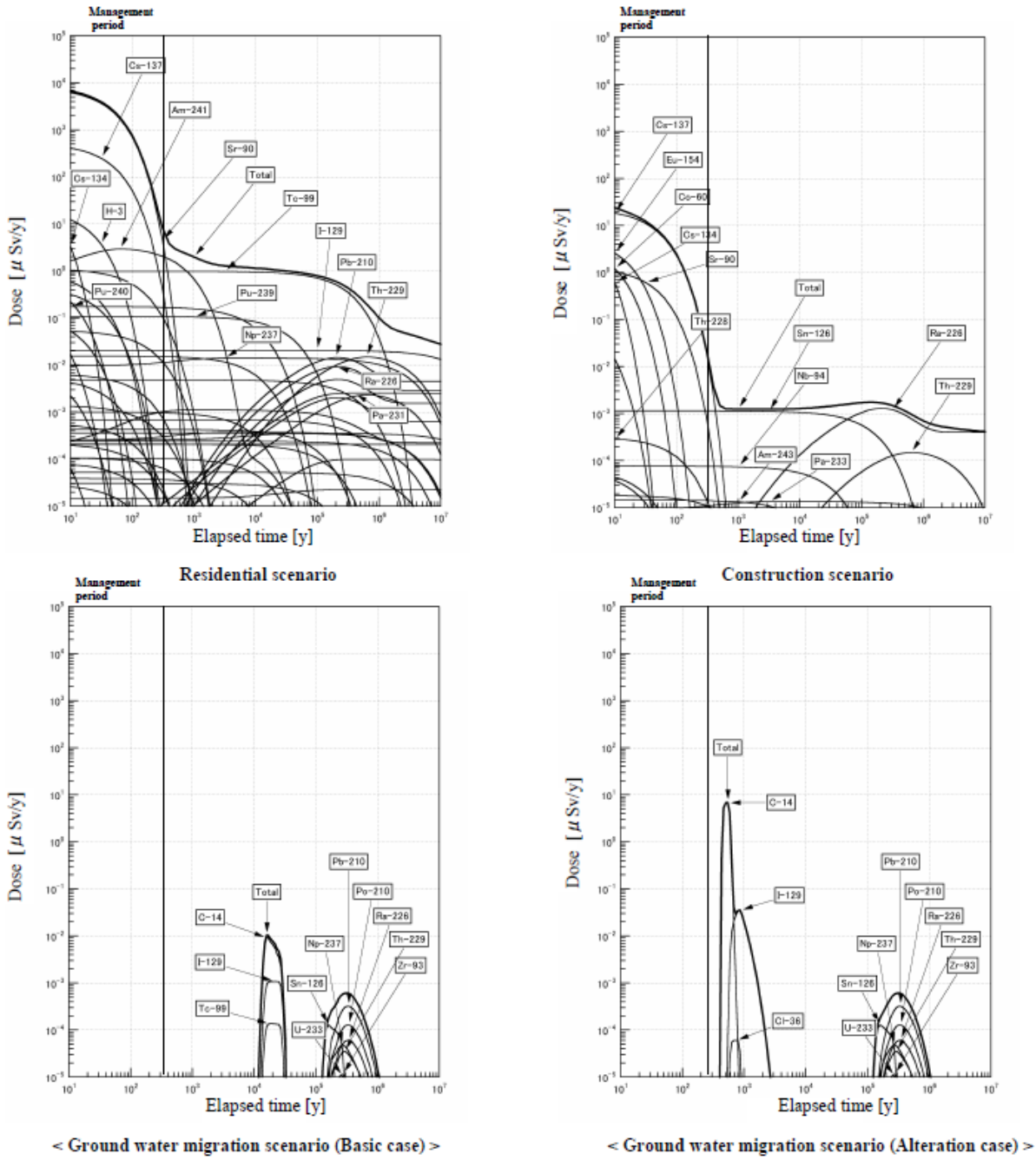

Figure 5.4.1-1 Dose assessments for concrete vault disposal

5.4.2 Safety assessment for intermediate-depth disposal

The results of dose assessments based on the assessment scenario, evaluation pathways and assessment parameters are summarized in Table 5.4.2-1 and Figures 5.4.2-1 - 5.4.2-2.

The longest paths of the 4 routes evaluated are the channel for river water use and river bank construction route. The radiation dose in the variation case is increased compared to that of the base case. The nuclides that dominate dose are C-14, Cl-36 and I-129, which migrate easily underground. TRU nuclides (and their daughters) become the dominant nuclides in the river bank construction route and the river bank residential route. However, the doses are small. Even if the sorption performance of C-14 is not high, it is considered that safety will be ensured because the leach rate of metals is small under repository conditions and the transmissivity of the facility for groundwater flow is decreased by bentonite in the EBS.

It is considered that an evaluation of facility and engineered barrier design should include assessments of the transport properties of C-14 and closure performance with respect to C-14. Additionally, considering that the peak dose is attained after several 1,000 years, the reliability of nuclide data should be improved. A long-term assessment of the behavior of the engineered barriers is necessary.

The assumed structure of the engineered barriers used for intermediate-depth disposal and the nuclides that control dose are the same as in the case of a deep geological repository. Therefore, it is appropriate to use evaluation results obtained for a deep geological repository.

Table 5.4.2-1 Results of dose assessments for intermediate-depth disposal

Evaluation pathway		Dose (μSv/y)	Time (y)	Dominant nuclide
Channel for river water use	Base case	2.3×10^{-2}	7.9×10^{4}	Cl-36, I-129
	Variation case	6.7×10^{0}	5.0×10^{3}	C-14
River bank construction route	Base case	3.1×10^{-4}	1.1×10^{6}	Th-229,Zr-93
	Variation case	1.6×10^{-5}	8.6×10^{6}	Th-229, Th-230
River bank residential route	Base case	5.0×10^{-4}	1.1×10^{6}	Th-229, Ra-226
	Variation case	4.1×10^{-5}	5.4×10^{3}	Cl-36, C-14
River bank agriculture route	Base case	6.3×10^{-2}	7.9×10^{4}	Cl-36
	Variation case	5.8×10^{-2}	5.4×10^{3}	Cl-36, C-14

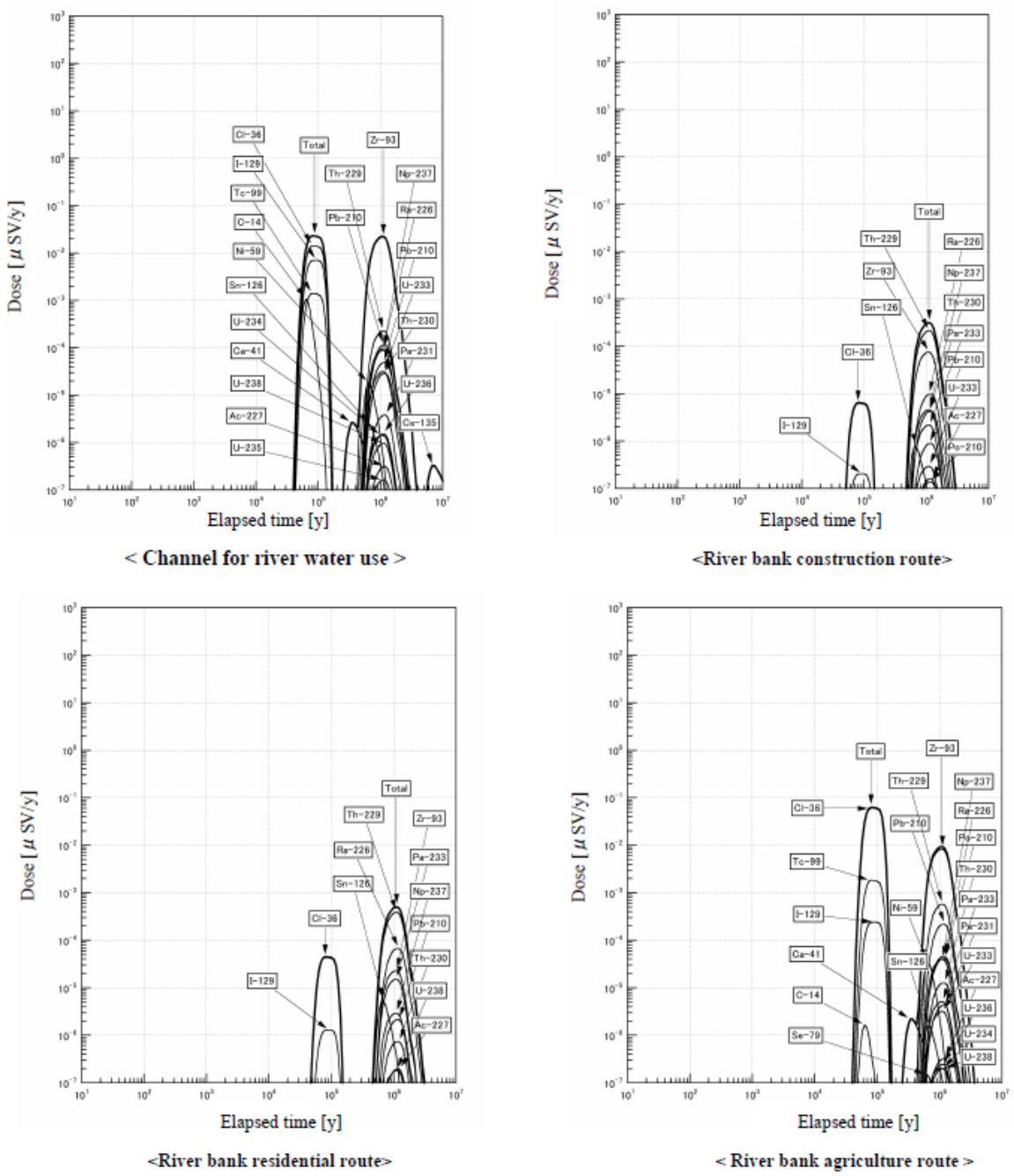

Figure 5.4.2-1 Dose assessments for intermediate-depth disposal (base case)

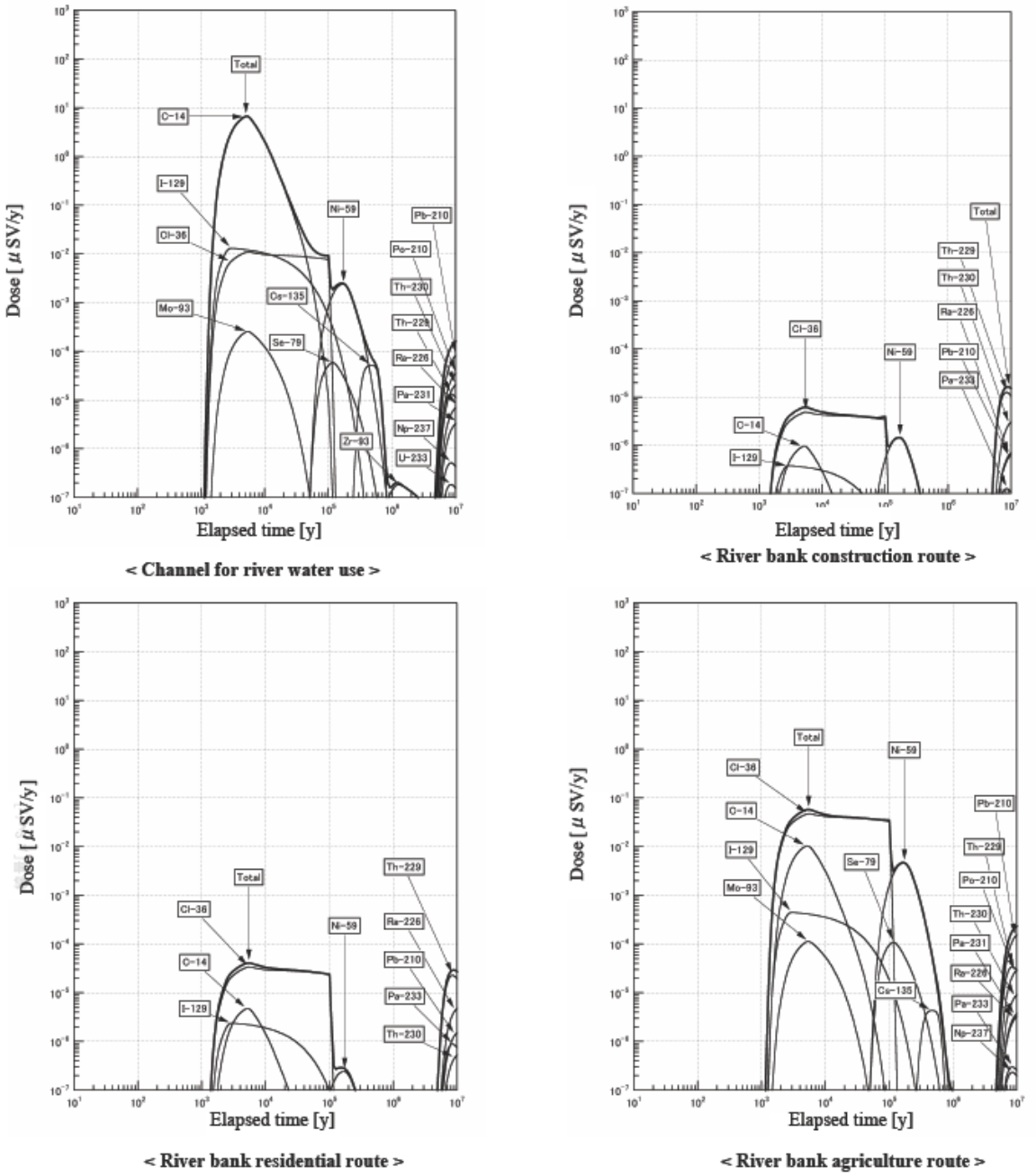

Figure 5.4.2-2 Dose assessments for intermediate-depth disposal (variation case)

5.5 Summary

Safety assessments were carried out using a revised waste database based on the current predicted volumes of waste to be generated. The assessment also used the disposal outline and safety assessment methodology in reports already published by the Nuclear Safety Commission of Japan (NSC, 1986, 1992, 2000a) for waste for concrete vault disposal and intermediate-depth disposal. The main results are as follows:

- The results of the dose assessment in the base case are estimated to be 10^{-4} to 10 μSv/y for concrete vault disposal and intermediate-depth disposal. Therefore safety can be assured.
- In intermediate-depth disposal, as in deep geological disposal, mobile nuclides such as C-14 dominate radiation doses.
- In the variation case which considers the range of nuclide parameters in dose assessment, safety is ensured by decreasing the leach rate or transmissivity with respect to inflowing water.

5.6 Future issues

- To evaluate the design of facilities and engineered barriers, post-closure performance and transport properties of nuclides, data with improved reliability and parameters relevant to specific geological conditions are required.
- Considering that the dose peak appears after several 1,000 years, confidence in long-term performance assessment needs to be improved. Hence, the results from research related to deep geological disposal systems should be used. These results are relevant to the assessment of the long-term behavior of engineered barriers, application of risk theory and the evaluation of uncertainty.
- It is necessary to consider the effects of nitrate salts and organic material in geological repository assessments, repository designs and repository safety assessments.

References

AEC (1985): On the Treatment and Disposal Policy of Radioactive Waste, Atomic Energy Commission of Japan. [written in Japanese]

AEC (1998): On the Basic Approach for Disposal of Low-Level Radioactive Waste that Exceeds the Present Ordinance Concentration Limit, Atomic Energy Commission of Japan, October 16th, 1998. [written in Japanese]

IAEA (1985): Generic of Models and Parameters for Assessing the Environmental Transfer of Radionuclides from Routine Releases, IAEA Safety Series No. 57.

IAEA (1994): Handbook of Parameter Values for the Prediction of Radionuclide Transfer in Temperate Environments, IAEA Technical Reports Series No. 364.

ICRP (1995): ICRP Publication 72: Age-dependent Doses to Members of the Public from Intake of Radionuclides: Part 5 Compilation of Ingestion and Inhalation Dose Coefficient.

Napier, B.A., Kennedy, W.E. Jr., et al. (1980): Assessment of Effectiveness of Geologic Isolation Systems, PNL-3209.

NSC (1985): Basic Concept for Safety Regulations for Land Disposal of Low-level Solid Radioactive Wastes, The Committee on Safety Regulations of Radioactive Waste, Nuclear Safety Commission of Japan, October, 1985. [written in Japanese]

NSC (1986): Reference Radionuclide Concentration Values for Low-level Radioactive Solid Waste to be Land Disposed (Interim Report), Nuclear Safety Commission of Japan, December, 1986. [written in Japanese]

NSC (1992): Reference Radionuclide Concentration Values for Low-level Radioactive Solid Waste to be Land Disposed (2nd Interim Report), Nuclear Safety Commission of Japan, June, 1992. [written in Japanese]

NSC (2000a): Reference Radionuclide Concentration Values for Low-level Radioactive Solid Waste to be Land Disposed (3rd Interim Report), Nuclear Safety Commission of Japan, June, 2000. [written in Japanese]

NSC (ed.) (2000b): White Paper on the Safety of Nuclear Energy in 1999, Printing Department of the Ministry of Finance. [written in Japanese]

Radioactive Waste Management Funding and Research Center (2002): Study on nuclide migration behavior of radioactivation materials (Corrosion rate test under an alkali and reducing atmosphere). [written in Japanese]