



JAEA-Review 2007-010
FEPC TRU-TR2-2007-01

FEPC

Second Progress Report on Research and Development for TRU Waste Disposal in Japan

**— Repository Design, Safety Assessment and Means
of Implementation in the Generic Phase —**

March 2007

Japan Atomic Energy Agency

The Federation of Electric Power Companies of Japan

本レポートは日本原子力研究開発機構と電気事業連合会が発行する成果報告書です。本レポートの入手並びに著作権利用に関するお問い合わせは、日本原子力研究開発機構¹⁾、もしくは電気事業連合会²⁾にてお問い合わせ下さい。なお、本レポートの全文は日本原子力研究開発機構ホームページ (<http://www.jaea.go.jp/index.shtml>) より発信されています。このほか財団法人原子力弘済会資料センター³⁾では実費による複写頒布を行っております。

- 1) 〒319-1195 茨城県那珂郡東海村白方白根2番地4
日本原子力研究開発機構 研究技術情報部 研究技術情報課
電話 029-282-6387, Fax 029-282-5920
- 2) 〒100-8118 東京都千代田区大手町1-9-4 経団連会館ビル
電気事業連合会 原子力部
電話 03-3279-2187
- 3) 〒319-1195 茨城県那珂郡東海村白方白根2番地4 日本原子力研究開発機構内

This report is issued by Japan Atomic Energy Agency (JAEA) and The Federation of Electric Power Companies of Japan (FEPC).
Inquiries about availability and/or copyright of this report should be addressed to JAEA¹⁾ or FEPC²⁾:

- 1) Intellectual Resources Section, Intellectual Resources Department, Japan Atomic Energy Agency
2-4 Shirakata Shirane, Tokai-mura, Naka-gun, Ibaraki-ken 319-1195, Japan
Tel +81-29-282-6387, Fax +81-29-282-5920
- 2) Nuclear Power Department, The Federation of Electric Power Companies of Japan
Keidanren-kaikan 9-4, 1-chome, Ohte-machi, Chiyoda-ku, Tokyo 100-8118, Japan
Tel +81-3-3279-2187

© Japan Atomic Energy Agency, 2007

© The Federation of Electric Power Companies of Japan, 2007

Contents

Chapter 1 Introduction

1.1	Context of the report	1-1
1.1.1	Status of studies on TRU waste disposal in Japan	1-1
1.1.2	Objective of this report	1-5
1.1.3	Approach to preparing the 2nd TRU progress report	1-6
1.2	Safety assurance for a geological repository	1-9
1.2.1	Background	1-9
1.2.2	Safety requirements	1-10
1.2.3	Safety standards	1-10
1.3	Geological environment	1-13
1.3.1	Principles for selecting the geological environment	1-13
1.3.1.1	Geological disposal	1-13
1.3.1.2	Concrete vault disposal and intermediate-depth disposal	1-13
1.3.2	Range of geological environments and the reference case	1-14
1.3.2.1	Geological repository	1-14
1.3.2.2	Concrete vault disposal and intermediate-depth disposal	1-18
1.3.3	Summary	1-18
	List of references	1-20

Chapter 2 Generation and characteristics of TRU waste

2.1	Types and management of TRU waste	2-1
2.1.1	Facilities which produce waste	2-1
2.1.2	Types of waste	2-1
2.1.3	Generation and management of TRU waste	2-7
2.2	Projected waste volumes	2-9
2.2.1	Waste treatment and packaging	2-9
2.2.2	Volumes of waste generated	2-12
2.3	Concentrations of radioactive materials in waste packages	2-22
2.4	Waste package classification	2-30
2.4.1	Principles for waste package classification	2-30
2.4.2	Type, quantity and characteristics of waste packages	2-32
2.5	Characteristics of waste packages	2-35
2.5.1	Contents of waste packages	2-35
2.5.2	Heat generated by waste packages	2-39
2.5.3	Classifying waste based on waste package characteristics	2-40

2.5.4	Classification of waste packages for deep geological disposal	2-43
2.5.4.1	Classification concept	2-43
2.5.4.2	Results of classifying waste packages for geological disposal	2-44
2.6	Summary	2-47
2.7	Future tasks	2-47
	List of references	2-48

Chapter 3 Repository design and engineering technology

3.1	Basic design concept	3-1
3.1.1	Basic construction technique for the engineered barriers and the disposal facility	3-3
3.1.1.1	Engineered barriers	3-3
3.1.1.2	Disposal facility	3-6
3.1.2	Design conditions	3-6
3.1.2.1	Waste	3-6
3.1.2.2	Disposal site and geological environment conditions	3-7
3.2	Design of the engineered barriers and the disposal facility	3-7
3.2.1	Design of engineered barriers	3-7
3.2.1.1	Design of waste package	3-7
3.2.1.2	Buffer material	3-9
3.2.1.3	Material design of filler	3-18
3.2.2	Design of the underground facility	3-20
3.2.2.1	Mechanical stability of the disposal tunnel	3-20
3.2.2.2	Layout of the underground facility	3-30
3.2.2.3	Design of backfill and plugs	3-46
3.3	Long-term mechanical stability of the near-field	3-51
3.3.1	Long-term mechanical behavior in the near-field	3-51
3.3.1.1	Factors affecting long-term mechanical behaviour	3-51
3.3.1.2	Evaluation method for long-term mechanical behaviour	3-52
3.3.2	Evaluation of long-term mechanical stability in the near-field	3-55
3.3.2.1	Long-term creep deformation of the host rock	3-55
3.3.2.2	Changes in engineered barrier properties and effects of swelling pressure of the buffer material	3-59
3.3.2.3	Influences of thermal stress due to thermal output of the waste	3-66
3.3.2.4	Effects of increasing pore pressure due to gas generation	3-67
3.3.2.5	Effect of extrusion of buffer material	3-69
3.3.3	Summary	3-69
3.3.4	Future tasks	3-70
3.4	Construction, operation and closure of the disposal facility	3-72

3.4.1	Structure of the disposal facility	3-72
3.4.1.1	Components of the disposal facility (surface and underground facilities)	3-72
3.4.1.2	Schedule for geological disposal of TRU waste	3-73
3.4.2	Construction	3-74
3.4.2.1	Surface facility	3-74
3.4.2.2	Underground facility	3-78
3.4.3	Operation	3-96
3.4.3.1	Evaluation of general operations	3-96
3.4.3.2	Transport and emplacement of waste	3-105
3.4.4	Closure	3-120
3.4.4.1	Construction of backfill material	3-120
3.4.4.2	Construction of plugs	3-124
3.4.4.3	Grouting	3-125
3.4.5	Management of the disposal facility	3-125
3.5	Summary	3-128
	List of references	3-130

Chapter 4 Evaluating the safety of geological disposal

4.1	Procedure for evaluating the safety of geological disposal	4-1
4.1.1	Safety assessment system	4-1
4.1.2	Methods for evaluating the influence of uncertainties	4-5
4.2	Summary of initial conditions	4-7
4.2.1	Safety requirements	4-7
4.2.2	Geological environment conditions	4-7
4.2.3	Inventory	4-8
4.2.4	Specified conditions	4-11
4.3	Scenario development	4-13
4.3.1	Production of a comprehensive FEP list	4-13
4.3.2	FEP classification	4-16
4.3.3	FEP screening considered in safety assessment	4-17
4.3.4	Classification of scenarios	4-22
4.3.5	Scenario description	4-23
4.3.5.1	Base scenario	4-23
4.3.5.2	Perturbation scenarios	4-25
4.3.5.3	Isolation failure scenarios	4-25
4.3.6	Important repository environment conditions for safety assessment	4-26
4.4	Establishing conditions in the disposal environment for safety assessment	4-28
4.4.1	Chemical condition of groundwater	4-28

4.4.1.1	Reference composition of groundwater	4-28
4.4.1.2	Geochemical variations of groundwater caused by interaction with repository materials	4-28
4.4.2	Effect of alteration of the engineered barrier	4-29
4.4.2.1	Evaluation of long-term engineered barrier performance	4-29
4.4.2.2	Effect of cement-bentonite interactions on long-term performance	4-30
4.4.2.3	Uncertainty in current knowledge and treatment of uncertainty in the nuclide migration analysis	4-56
4.4.2.4	Summary	4-63
4.4.2.5	Future issues	4-63
4.4.3	Hyperalkaline alteration of host rock around the disposal facility	4-64
4.4.3.1	Types of surrounding host rock	4-64
4.4.3.2	Knowledge of the chemical alteration of surrounding host rocks by a high-pH plume	4-64
4.4.3.3	Impact analysis for hyperalkaline alteration of rocks surrounding a facility	4-66
4.4.3.4	Summary	4-71
4.4.3.5	Future issues	4-71
4.4.4	Hydraulic conditions of the near-field	4-72
4.4.4.1	Analytical assumptions	4-72
4.4.4.2	Model and data for hydrogeological analyses	4-73
4.4.4.3	Analytical results	4-79
4.4.4.4	Summary	4-80
4.4.4.5	Future issues	4-81
4.4.5	Effects of colloids	4-81
4.4.5.1	Effect of colloids in the EBS	4-81
4.4.5.2	Effects of colloids in the natural barrier	4-85
4.4.5.3	Summary	4-87
4.4.5.4	Future issues	4-87
4.4.6	Effects of organic materials	4-88
4.4.6.1	Organic materials considered	4-88
4.4.6.2	Effects of organic materials from waste	4-88
4.4.6.3	Effect of cement additives	4-90
4.4.6.4	Effects of natural organic material	4-91
4.4.6.5	Summary	4-91
4.4.6.6	Future issues	4-92
4.4.7	Effects of microbes	4-92
4.4.7.1	Microbial activity in a geological repository for TRU waste	4-94
4.4.7.2	Effect on nuclide solubility and sorption	4-95
4.4.7.3	Effects on the engineered and natural barriers	4-97
4.4.7.4	Effect on gas generation	4-98

4.4.7.5	Summary	4-98
4.4.7.6	Future issues	4-99
4.4.8	Effects on the radiation field	4-99
4.4.8.1	Radiation field in the EBS considering the shielding effect	4-99
4.4.8.2	Exposure damage to EBS material	4-103
4.4.8.3	Radiolysis of pore water	4-105
4.4.8.4	Summary	4-109
4.4.8.5	Future issues	4-110
4.4.9	Nitrate salt effects	4-110
4.4.9.1	Generation of nitrate with organic material inclusions and impact assessment	4-110
4.4.9.2	Chemical transition of nitrate ions	4-110
4.4.9.3	Effect of nitrate salt on cementitious material	4-111
4.4.9.4	Effect on solubility of radionuclides and sorption distribution coefficients	4-112
4.4.9.5	Summary	4-113
4.4.9.6	Future issues	4-114
4.4.10	Quantification of gas effects and the behavior of gaseous nuclides	4-115
4.4.10.1	Generation and migration of non-radiogenic gases	4-115
4.4.10.2	Generation and migration of radioactive gas	4-122
4.4.10.3	Summary	4-127
4.4.10.4	Future issues	4-127
4.5	Radionuclide transport analysis and dose assessment	4-132
4.5.1	Analytical cases	4-132
4.5.2	Analysis of the Reference Case	4-135
4.5.2.1	Assumptions in the analysis	4-135
4.5.2.2	Dataset for radionuclide transport analysis (RAMDA: <u>RA</u> dionuclides <u>M</u> igration <u>DA</u> taset)	4-137
4.5.2.3	Models and parameters of the engineered barriers	4-138
4.5.2.4	Radionuclide transport data and models in the host rock	4-151
4.5.2.5	Radionuclide transport analysis model in faults and associated data	4-157
4.5.2.6	Biosphere model and data	4-159
4.5.2.7	Safety assessment model chain	4-161
4.5.2.8	Analysis results	4-161
4.5.2.9	Summary	4-168
4.5.3	Analysis of alternative cases in the base scenario	4-168
4.5.3.1	Model expansion necessary for analyzing the alternative cases	4-169
4.5.3.2	Parameters used in alternative cases	4-170
4.5.3.3	Analytical results	4-178
4.5.3.4	Summary	4-187

4.5.3.5	Future issues	4-187
4.5.4	Evaluation of uncertainty in the base scenario	4-189
4.5.4.1	Comprehensive sensitivity analysis	4-189
4.5.4.2	Specification of parameter variation ranges	4-192
4.5.4.3	Analytical results	4-194
4.5.4.4	Summary	4-202
4.5.4.5	Future issues	4-203
4.5.5	Analysis of perturbation scenario	4-203
4.5.5.1	Natural event impact case	4-203
4.5.5.2	Sealing defects in the engineered barrier system	4-214
4.5.5.3	Future human intrusion case	4-216
4.5.5.4	Summary of perturbation scenarios	4-222
4.5.6	Analysis of isolation failure scenarios	4-224
4.5.6.1	Surface exposure of the disposal facility by uplift and erosion	4-224
4.5.6.2	Accidental penetration of the repository by drilling	4-227
4.6	Summary of disposal safety	4-233
4.6.1	Method for conducting safety assessments of a geological repository	4-233
4.6.2	Summary of preconditions used in the safety assessment	4-233
4.6.3	Scenario development	4-234
4.6.4	Specification of environmental condition in the safety assessment	4-234
4.6.5	Radionuclide transport analysis and dose assessments	4-234
4.6.5.1	Specification of analytical cases	4-234
4.6.5.2	Results of safety assessment for the base scenario of the groundwater scenario	4-234
4.6.5.3	Results of the safety assessment of the perturbation scenario in the groundwater scenario	4-235
4.6.5.4	Results of safety assessment for the isolation failure scenario	4-236
4.6.6	Summary of safety	4-237
	List of references	4-238

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	5-1
5.2	Future plans for safety assessment of concrete vault disposal and intermediate-depth disposal	5-4
5.3	Scenario evaluation for concrete vault disposal and intermediate-depth disposal	5-5
5.3.1	Assessment scenarios and models for concrete vault disposal	5-5
5.3.2	Assessment parameters for concrete vault disposal	5-5
5.3.3	Assessment scenarios and models for intermediate-depth disposal	5-6

5.3.4	Assessment parameters for intermediate-depth disposal	5-6
5.4	Safety assessments of concrete vault disposal and intermediate-depth disposal	5-8
5.4.1	Safety assessment of concrete vault disposal	5-8
5.4.2	Safety assessment for intermediate-depth disposal	5-10
5.5	Summary	5-13
5.6	Future issues	5-13
	List of references	5-14

Chapter 6 Evaluation aimed at optimizing waste disposal

6.1	Basic concept for rationalization	6-1
6.2	Evaluation of co-location disposal with high-level waste	6-2
6.2.1	Disposal conditions specified in other countries	6-2
6.2.2	Evaluation of the influences of interactions	6-6
6.2.2.1	Elicitation and classification of influencing factors	6-6
6.2.2.2	Thermal effects	6-8
6.2.2.3	Effects of organic material	6-12
6.2.2.4	Effects of nitrate	6-20
6.2.2.5	Effects of high pH	6-31
6.2.3	Evaluation of co-location disposal	6-47
6.2.4	Summary	6-48
6.2.5	Future issues	6-49
6.3	Effects of waste classification by α -emitting nuclide concentration at intermediate-depth disposal	6-52
6.3.1	Need to evaluate the classification concept for intermediate-depth disposal	6-52
6.3.2	Upper concentration limit	6-52
6.3.3	Upper limit of α -emitting nuclide concentration	6-54
6.3.4	Amounts of waste generated	6-57
6.3.5	Safety assessment for intermediate-depth disposal	6-61
6.3.6	Safety assessment of geological disposal	6-64
6.3.7	Summary	6-64
6.4	Effect of return method on waste returned from overseas	6-64
6.4.1	Summary of returned radioactive waste	6-64
6.4.1.1	Summary of waste	6-64
6.4.1.2	Waste treatment and waste packages	6-66
6.4.1.3	Quantities of waste generated	6-66
6.4.1.4	Properties of waste packages	6-70
6.4.1.5	Waste classification	6-73
6.4.2	Disposal method	6-77

6.4.3	Safety assessment of concrete vault disposal	6-77
6.4.4	Safety assessment of geological disposal	6-77
6.4.5	Summary	6-78
6.4.6	Future issues	6-79
	List of references	6-80

Chapter 7 Evaluation of alternative technologies for the TRU waste disposal facility

7.1	Basic concepts for alternative technologies	7-1
7.1.1	Alternative engineered barrier technologies	7-2
7.1.1.1	Iodine immobilization technologies	7-2
7.1.1.2	Long-term C-14 confinement technologies	7-3
7.1.2	Alternative approaches for reducing remaining uncertainties	7-3
7.1.2.1	Development of low-alkaline cement	7-3
7.1.2.2	Development of nitrate decomposition technologies	7-4
7.2	Iodine immobilization technologies	7-5
7.3	Long-term C-14 confinement technology	7-9
7.4	Development of low-alkaline cement	7-13
7.4.1	Adjusting the composition of raw materials and cement additives	7-13
7.4.2	Adding large quantities of pozzolans	7-15
7.5	Development of nitrate decomposition technologies	7-17
	List of references	7-19

Chapter 8 Conclusions and future R&D requirements

8.1	Conclusions	8-1
8.2	Future prospects for repository implementation	8-5
8.3	Future R&D requirements	8-7
	List of references	8-16

Appendices

Relevant to Chapter 1

1A Framework for the 2nd TRU progress report

Relevant to Chapter 2

2A Effect of 'burn-up' of radioactive material in spent fuel

2B Weight of radioactive substances and content for each waste group

Relevant to Chapter 3

3A Alternative concepts for waste packaging

3B Envisaged specifications of cementitious material for structural analysis

Relevant to Chapter 4

4A Input data for alternative analytical cases

Relevant to Chapter 5

5A Assessment scenario and model for concrete vault disposal

5B Assessment parameters for concrete vault disposal

5C Assessment scenario and model for intermediate-depth disposal

5D Assessment parameters for intermediate-depth disposal

Figure list

Figure 2.1.2-1	Reprocessing and generation of radioactive waste	2-6
Figure 2.2.1-1	Waste configuration (domestic)	2-9
Figure 2.2.2-1	Flowchart showing the procedure for estimating volumes of radioactive materials (concentrations) and numbers of waste packages	2-13
Figure 2.3-1	Distributions of radioactivity in TRU wastes from different sources	2-25
Figure 2.5.2-1	Change with time in heat production in a waste package containing commercial hulls and ends	2-40
Figure 3.1-1	Assessment flow for the engineered barriers and disposal facility	3-2
Figure 3.1.1.1-1	Basic concept for the engineered barrier system	3-4
Figure 3.1.1.2-1	Basic concept for the TRU repository	3-6
Figure 3.2.1.2-1	Peclet number for buffer material for vertical flow (mean flow velocity of groundwater)	3-10
Figure 3.2.1.2-2	Peclet number for buffer material for horizontal flow (mean flow velocity of groundwater)	3-11
Figure 3.2.1.2-3	Decreasing effect on release rates with increasing thickness of the buffer material ...	3-13
Figure 3.2.1.2-4	Relationship between effective clay dry density and hydraulic conductivity for various types of bentonite	3-14
Figure 3.2.1.2-5	Relationship between effective clay dry density and hydraulic conductivity for Ca bentonite	3-15
Figure 3.2.1.2-6	Relationship between effective clay dry density and hydraulic conductivity for fresh and saline water	3-16
Figure 3.2.2-1	Evaluation process for the layout of the underground facility	3-20
Figure 3.2.2.1-1	Example for analytical model	3-22
Figure 3.2.2.1-2	Analytical steps	3-22
Figure 3.2.2.1-3	Distribution of local safety factors	3-23
Figure 3.2.2.1-4	Distribution of support stress	3-23
Figure 3.2.2.1-5	Example of analytical model (horseshoe-shape)	3-25
Figure 3.2.2.1-6	Analytical steps	3-25
Figure 3.2.2.1-7	Distribution of maximum shearing strain	3-26
Figure 3.2.2.1-8	Analytical model example	3-27
Figure 3.2.2.1-9	Seismic velocity structure in a bedrock model	3-27
Figure 3.2.2.1-10	Distribution of local safety factors	3-28
Figure 3.2.2.1-11	Temporal variation of support stress	3-28
Figure 3.2.2.2-1	Shape of steel structural framework (example)	3-31
Figure 3.2.2.2-2	Shape of concrete support (example)	3-31

Figure 3.2.2.2-3	Cross-sections of disposal tunnels used in the thermal analysis	3-34
Figure 3.2.2.2-4	Example of thermal analysis model	3-35
Figure 3.2.2.2-5	Locations of points used in evaluating thermal histories	3-36
Figure 3.2.2.2-6	Cross-sections showing number of emplaced waste packages (example)	3-38
Figure 3.2.2.2-7	Cross-section views of disposal tunnels for soft rock reference case (SR-C, depth 500 m, circular cross-section)	3-40
Figure 3.2.2.2-8	Example reference case layout in soft rock (SR-C, depth 500 m, circular cross-section)	3-41
Figure 3.2.2.2-9	Cross-section views of disposal tunnels for the hard rock reference case (HR, depth 1,000 m, circular cross-section)	3-42
Figure 3.2.2.2-10	Example reference case layout in hard rock (HR, depth 1,000 m, circular cross-section)	3-43
Figure 3.2.2.2-11	Cross-section of horseshoe-shaped disposal tunnel in hard rock (HR, depth 1,000 m)	3-44
Figure 3.2.2.2-12	Example layout of horseshoe-shaped disposal tunnels in hard rock (HR, depth 1,000 m)	3-45
Figure 3.2.2.2-13	Example layout in soft rock with Group 3 waste arranged parallel to groundwater flow (SR-C, depth 500 m, circular cross-section)	3-46
Figure 3.2.2.3-1	Function of backfill and plugs in a disposal facility for TRU waste	3-47
Figure 3.2.2.3-2	Closure concept for circular disposal tunnel	3-50
Figure 3.2.2.3-3	Closure concept for horseshoe-shaped disposal tunnel	3-50
Figure 3.2.2.3-4	Closure concept for access tunnel	3-50
Figure 3.3-1	Process influence diagram	3-52
Figure 3.3-2	Result of creep deformation analysis (Case 2)	3-58
Figure 3.3-3	Creep displacement in the central apex after 1 million years	3-58
Figure 3.3-4	Analytical flow	3-59
Figure 3.3-5	Concept of deformation behaviour of cement material	3-62
Figure 3.3-6	Concept of swelling behaviour of bentonite materials	3-62
Figure 3.3-7	Variation in mechanical properties of mortar and concrete	3-63
Figure 3.3-8	Swelling behaviour of buffer material during unloading	3-64
Figure 3.3-9	Variation in equilibrium swelling pressure of buffer	3-64
Figure 3.3-10	Deformation contours	3-65
Figure 3.3-11	Thermal stress analysis	3-67
Figure 3.3-12	Local safety factor in sedimentary rocks surrounding disposal tunnels for Group 2 waste	3-68
Figure 3.4.1.2-1	Disposal schedule for TRU waste (tentative)	3-74
Figure 3.4.2.2-1	Concept for underground facility	3-78
Figure 3.4.2.2-2	Cross-section of a vertical shaft	3-79

Figure 3.4.2.2-3	Cross-section of inclined shaft	3-80
Figure 3.4.2.2-4	Cross-section of main tunnel and connecting tunnel	3-81
Figure 3.4.2.2-5	Cross-section of a circular disposal tunnel in soft rock	3-82
Figure 3.4.2.2-6	Cross-section of circular and horseshoe-shaped disposal tunnels in hard rock	3-82
Figure 3.4.2.2-7	Concept for the excavation process of large disposal tunnels	3-83
Figure 3.4.2.2-8	Construction of buffer material in the circular disposal tunnel	3-91
Figure 3.4.2.2-9	Construction of buffer material in horseshoe-shaped disposal tunnel	3-92
Figure 3.4.2.2-10	Construction of steel structural framework (circular disposal tunnel, Groups 1 and 2)	3-93
Figure 3.4.2.2-11	Construction of reinforced concrete structural framework (horseshoe-shaped disposal tunnel, Groups 1 and 2)	3-94
Figure 3.4.2.2-12	Construction of reinforced concrete structural framework (horseshoe-shaped disposal tunnel, Groups 3 and 4)	3-94
Figure 3.4.3.1-1	Basic flow of TRU waste disposal	3-97
Figure 3.4.3.1-2	Flow for waste emplacement in circular disposal tunnels and construction of filling material and buffer material	3-99
Figure 3.4.3.1-3	Flow for waste emplacement in horseshoe-shaped disposal tunnels and construction of filling material and buffer material	3-100
Figure 3.4.3.1-4	Operations management for a TRU waste repository	3-104
Figure 3.4.3.2-1	Shape of inclined shaft	3-106
Figure 3.4.3.2-2	Shape of the main tunnel	3-107
Figure 3.4.3.2-3	Use of host rock as crane foundation	3-114
Figure 3.4.3.2-4	Overall view of handling equipment for underground facility (circular disposal tunnel)	3-118
Figure 3.4.3.2-5	Overall view of handling equipment for underground facility (horseshoe-shaped disposal tunnel)	3-119
Figure 3.4.4.1-1	Example of construction of backfill in the upper part of a disposal tunnel (circular disposal tunnel)	3-122
Figure 3.4.4.1-2	Example of construction of backfill in the upper part of a disposal tunnel (horseshoe-shaped disposal tunnel)	3-123
Figure 4.1.1-1	Safety assessment system	4-2
Figure 4.1.2-1	Characteristics of the comprehensive sensitivity analysis	4-5
Figure 4.2.3-1	Temporal variation in radioactivity	4-9
Figure 4.2.4-1	Cross-sections of disposal tunnels specified as a reference	4-12
Figure 4.3-1	Flow of scenario development	4-13
Figure 4.3.4-1	Classification of scenarios	4-23
Figure 4.3.6-1	Hierarchy of phenomena connected with safety assessment	4-26
Figure 4.4.2.2-1	Variation in the pH of the cement leachate	4-32

Figure 4.4.2.2-2	Evaluation result for chemical environment in disposal facility without bentonite (FRHP–with fractured cement)	4–40
Figure 4.4.2.2-3	Outline of the mineralogical consequences of aluminosilicate mineral	4–48
Figure 4.4.2.2-4	Multiple alteration scenarios	4–49
Figure 4.4.2.2-5	One dimensional analysis model (The mesh increments are also shown.)	4–51
Figure 4.4.2.2-6	Temporal variation of pH distribution	4–53
Figure 4.4.2.2-7	Temporal variation of porosity distribution	4–53
Figure 4.4.2.2-8	Calculated mineral distribution (after 100,000 years)	4–54
Figure 4.4.2.3-1	Mineral distribution in the case that assumes that stable zeolites and K-feldspar are not formed	4–58
Figure 4.4.2.3-2	Time change of pH distribution in the case assuming that stable zeolites and K-feldspar are not formed	4–58
Figure 4.4.2.3-3	Mineral distribution in the case that assumes that the porosity does not change (after 100,000 years)	4–61
Figure 4.4.3-1	System for the analysis of hyperalkaline alteration in the surrounding host rock	4–67
Figure 4.4.3-2	Predicted distribution of solutes in the liquid phase after 10,000 years (freshwater type groundwater, crystalline rock)	4–69
Figure 4.4.3-3	Predicted variations in the volumetric fractions of solid phases after 10,000 year (freshwater type groundwater, crystalline rock)	4–70
Figure 4.4.3-4	Variation of pH (freshwater type groundwater, crystalline rock)	4–70
Figure 4.4.3-5	Spatio-temporal variation of pH and porosity (freshwater type groundwater, crystalline rock)	4–71
Figure 4.4.4-1	Relationship between porosity and the hydraulic conductivity in hardened cement paste (triangle) and mortar (diamond)	4–75
Figure 4.4.4-2	Relationship between the groundwater flux and the increase in hydraulic conductivity of the region affected by excavation	4–77
Figure 4.4.4-3	Geometry for the hydraulic analyses	4–78
Figure 4.4.4-4	Relationship between Darcy flow velocity of cement mortar, buffer material and the hydraulic conductivity of buffer material	4–80
Figure 4.4.5-1	Filtration effects of buffer on colloids	4–82
Figure 4.4.5-2	Relationship between colloid mass concentration and ionic strength of cement pore water and groundwater	4–83
Figure 4.4.5-3	Relationship between colloid mass concentration and decrease in distribution coefficient	4–85
Figure 4.4.8-1	Model system used in shielding calculations	4–100
Figure 4.4.8-2	Air-kerma rate in the model system	4–102
Figure 4.4.8-3	Radiation absorption rate and adsorption dose in cement mortar (part adjacent to canister)	4–104

Figure 4.4.8-4	Radiation absorption rate and adsorption dose in buffer material (inside boundary)	4-104
Figure 4.4.8-5	Generation rates of oxidizing agents and the supply rates of reducing nuclides in the canister	4-106
Figure 4.4.8-6	Cumulative amounts of oxidizing agents generated in the canister and reducing volume of the canister	4-106
Figure 4.4.8-7	Concentrations of oxidizing agent generated by γ -ray penetration of mortar and buffer material	4-109
Figure 4.4.10.1-1	Maximum amounts of gas generated and cumulative amounts of generated gas –Crystalline bedrock	4-118
Figure 4.4.10.1-2	Variation of gas generation rates with time –Crystalline bedrock, Group 2 (canister)	4-119
Figure 4.4.10.1-3	Example results from a gas migration analysis –Crystalline bedrock, Group 2 (canister)	4-122
Figure 4.4.10.2-1	$^{14}\text{CH}_4$ generation rate –Crystalline bedrock, Group 2 (canister)	4-125
Figure 4.4.10.2-2	$^{14}\text{CH}_4$ generation rate –Crystalline bedrock (Group 3: 200L drum)	4-125
Figure 4.4.10.2-3	Migration rate of $^{14}\text{CH}_4$ on the surface (Group 3)	4-126
Figure 4.5.2-1	Radionuclide transport pathway in the Reference Case	4-136
Figure 4.5.2-2	One-dimensional concept for the model for radionuclide transport analysis in the engineered barriers (example of Group 2 waste)	4-139
Figure 4.5.2-3	Conceptual illustration of the one-dimensional parallel-plate model used in radionuclide transport analysis in the natural barrier	4-152
Figure 4.5.2-4	Model chain used in the safety assessment and associated flow of information for TRU waste disposal	4-161
Figure 4.5.2-5	Radionuclide release rates from engineered barriers for each waste group	4-162
Figure 4.5.2-6	Release rate of each radionuclide from the engineered barriers	4-162
Figure 4.5.2-7	Release rate from the host rock (for each waste group)	4-163
Figure 4.5.2-8	Release rates from the host rock (individual radionuclides)	4-164
Figure 4.5.2-9	Radionuclide release rates for each waste group to the biosphere via the fault	4-165
Figure 4.5.2-10	Release rates to the biosphere via the fault	4-165
Figure 4.5.2-11	Release rates from each barrier component	4-166
Figure 4.5.2-12	Results of dose assessment (for each waste group)	4-167
Figure 4.5.2-13	Results of dose assessment (for each radionuclide)	4-167
Figure 4.5.3-1	Comparison of the Reference Case with alternative cases in the groundwater scenario	4-180
Figure 4.5.3-2	Comparison of results of Reference Case with alternative cases that use hypothetical parameters	4-181
Figure 4.5.3-3	Analytical results for the geological environment alteration case	4-182

Figure 4.5.3-4	Analytical results for the log-mean of fracture transmissivity	4-183
Figure 4.5.3-5	Analytical results for transmissivity	4-184
Figure 4.5.3-6	Analytical results for the host rock matrix diffusion depth	4-184
Figure 4.5.3-7	Analytical results for proportion of fracture surface from which radionuclides can diffuse into the matrix	4-185
Figure 4.5.3-8	Analytical results for porosity of the host rock matrix	4-186
Figure 4.5.3-9	Analytical results for the length of faults	4-186
Figure 4.5.4-1	Identification of parameter importance	4-190
Figure 4.5.4-2	Depth of host rock matrix diffusion versus maximum dose (I-129 only)	4-191
Figure 4.5.4-3	Parameter ranges of sorption distribution coefficients (Kd) of cementitious filling materials used in the comprehensive sensitivity analysis	4-192
Figure 4.5.4-4	Results of the statistical analysis and comparisons between level of importance and influence on maximum dose	4-194
Figure 4.5.4-5	Relationship between log-mean of transmissivity and maximum dose	4-195
Figure 4.5.4-6	Statistical analysis results and comparisons between parameter importance and influence on maximum dose	4-196
Figure 4.5.4-7	Relationship between host rock matrix diffusion depth, host rock matrix sorption distribution coefficient and maximum dose	4-197
Figure 4.5.4-8	Confirming the successful condition	4-198
Figure 4.5.4-9	Successful condition below target dose and results of a statistical analysis	4-199
Figure 4.5.4-10	Results of a statistical analysis	4-200
Figure 4.5.4-11	Sensitivity of dose to nuclide leach time in Group 1	4-201
Figure 4.5.5-1	Geological environment in the uplift/erosion scenario	4-204
Figure 4.5.5-2	Evaluation of the uplift/erosion scenario	4-206
Figure 4.5.5-3	Schematic diagrams of geological environments in the climate/sea-level scenario	4-209
Figure 4.5.5-4	Evaluation of the climate/sea-level scenario (inland location)	4-213
Figure 4.5.5-5	Evaluation of the climate/sea-level scenario (coastal location)	4-214
Figure 4.5.5-6	Schematic diagram of the sealing defect scenario	4-214
Figure 4.5.5-7	Dose evaluations for the sealing defect scenario	4-216
Figure 4.5.5-8	Schematic diagram of the well drilling scenario (in aquifer)	4-217
Figure 4.5.5-9	Evaluation of the well drilling/water sampling scenario	4-219
Figure 4.5.5-10	Scenario for the creation of transport pathway by drilling	4-220
Figure 4.5.5-11	Evaluation of a new radionuclide pathway formed due to accidental drilling	4-222
Figure 4.5.5-12	Summary of the results of an evaluation of the perturbation scenarios	4-223
Figure 4.5.6-1	Conceptual model of surface exposure due to uplift and erosion	4-226
Figure 4.5.6-2	Surface exposure of the disposal facility (crystalline rock with an uplift/erosion rate of 1 mm/y)	4-226

Figure 4.5.6-3	Risk of accidental repository penetration by deep drilling	4-232
Figure 5.1-1	Outline of concrete vault disposal	5-1
Figure 5.1-2	Outline of intermediate-depth disposal	5-2
Figure 5.1-3	Outline of concrete vault disposal	5-3
Figure 5.4.1-1	Dose assessments for concrete vault disposal	5-9
Figure 5.4.2-1	Dose assessments for intermediate-depth disposal (base case)	5-11
Figure 5.4.2-2	Dose assessments for intermediate-depth disposal (variation case)	5-12
Figure 6.2.1-1	Concept for the disposal of HLW, spent fuel and TRU waste developed in Switzerland	6-5
Figure 6.2.2.2-1	Temporal changes in heat generation rates in vitrified waste and in waste packages containing hulls and ends	6-8
Figure 6.2.2.2-2	Heat conduction analysis model	6-9
Figure 6.2.2.2-3	Temperature change with respect to separation distance and time in crystalline rock	6-11
Figure 6.2.2.2-4	Temperature change with respect to separation distance and time in sedimentary rock	6-11
Figure 6.2.2.3-1	Chemical structures of cellulose and ISA	6-12
Figure 6.2.2.3-2	Illustration of the analytical model	6-13
Figure 6.2.2.3-3	Results of the simulation of an organic plume in crystalline rock (hydraulic conductivity $1 \times 10^{-9} \text{ m s}^{-1}$)	6-16
Figure 6.2.2.3-4	Results of the simulation of an organic plume in sedimentary rock (hydraulic conductivity $1 \times 10^{-9} \text{ m s}^{-1}$)	6-17
Figure 6.2.2.3-5	Relationship between Pu solubility and ISA concentration	6-18
Figure 6.2.2.3-6	Relationships between distribution coefficients of Eu and Th and ISA concentrations	6-18
Figure 6.2.2.4-1	Relationship between sorption distribution coefficient of Γ^- on rock and Na^+ concentration in liquid	6-22
Figure 6.2.2.4-2	Analytical model	6-23
Figure 6.2.2.4-3	Temporal variations of nitrate distribution (crystalline bedrock: hydraulic conductivity of host rock $1 \times 10^{-9} \text{ m s}^{-1}$)	6-24
Figure 6.2.2.4-4	Temporal variations of nitrate distribution (sedimentary rock: hydraulic conductivity of host rock $1 \times 10^{-9} \text{ m s}^{-1}$)	6-25
Figure 6.2.2.4-5	Results of X-ray diffraction analyses of bentonite before and after alteration tests ...	6-28
Figure 6.2.2.4-6	Permeability test using sodium nitrate solution (3 mol/dm^3)	6-29
Figure 6.2.2.4-7	Relationship between the sorption distribution coefficient of Cs^+ on rock and minerals and the Na^+ concentration of liquid	6-30
Figure 6.2.2.5-1	Illustration of the modeled regions	6-32
Figure 6.2.2.5-2	Distribution of solute concentrations in the liquid phase after 10,000 years (freshwater	

	type groundwater, crystalline rock)	6–37
Figure 6.2.2.5-3	Spatial variations in the volumetric fractions of solid phases after 10,000 years (freshwater type groundwater, crystalline rock)	6–38
Figure 6.2.2.5-4	Distribution of solute concentrations in the liquid phase after 10,000 years (seawater type groundwater, crystalline rock)	6–39
Figure 6.2.2.5-5	Spatial variations in the volumetric fractions of solid phases after 10,000 years (seawater type groundwater, crystalline rock)	6–40
Figure 6.2.2.5-6	Variation in pH (1×10^{-9} m s ⁻¹ , freshwater type groundwater, crystalline rock)	6–41
Figure 6.2.2.5-7	Variation in pH (1×10^{-9} m s ⁻¹ , seawater type groundwater, crystalline rock)	6–41
Figure 6.2.2.5-8	Temporal and spatial variations in pH	6–42
Figure 6.2.2.5-9	Results of hydraulic conductivity tests on bentonite	6–43
Figure 6.2.2.5-10	Diffusion coefficient of bentonite as a function of dry density	6–44
Figure 6.2.2.5-11	pH dependence of smectite dissolution rate	6–45
Figure 6.2.2.5-12	Conditions for passivation of carbon steel in buffer material	6–46
Figure 6.2.3-1	Example of the basic concept for co-location disposal of TRU waste and HLW	6–50
Figure 6.2.3-2	Co-location disposal concept of TRU waste and HLW	6–51
Figure 6.3.3-1	Concentrations of α -emitting nuclides in the main waste	6–57
Figure 6.3.5-1	Results of dose assessment for intermediate-depth disposal (100 GBq/t)	6–63
Figure 6.4.5-1	Optimization depending on the selected method for returning waste	6–78
Figure 7.2-1	Micrographs of AgI in rock	7–7
Figure 7.2-2	Relationship between amount of iodine and silicate leached and \sqrt{t} (leaching period) for long-term leaching experiments	7–8
Figure 7.2-3	Schematic diagram of iodine leaching in rock	7–8
Figure 7.3-1	Concrete packages for containment of C-14	7–12
Figure 7.3-2	Result of coupled water permeation-chemical degradation analysis	7–12
Figure 7.3-3	Passivation–depassivation transition of Ti-Gr.1 at 80°C	7–12
Figure 7.3-4	Initial corrosion sensitivity of titanium alloy in a TRU waste disposal environment	7–12
Figure 7.3-5	Result of SCC progress evaluation test of Ti alloy	7–12
Figure 7.4-1	Comparison of pH of leachates from LAC and OPC	7–14
Figure 7.4-2	Comparison of the compressive strength of LAC and OPC	7–14
Figure 7.4-3	pH variation of leachate from OPC and HFSC	7–15
Figure 7.4-4	Variation of Ca concentration of leachate from OPC and HFSC	7–15
Figure 7.5-1	Methods for decomposing and removing nitrates	7–17

Table list

Table 1.1-1	Status of discussions, safety regulations and safety standards for radioactive waste disposal	1-4
Table 1.2-1	Safety requirements in this report	1-12
Table 1.3-1	Specification of parameters for the geological environment for a repository	1-15
Table 1.3-2	Specification of geological environment conditions for shallow and intermediate-depth disposal	1-19
Table 2.1.2-1	Radioactive wastes generated during operation	2-4
Table 2.1.2-2	Radioactive waste generated by dismantling	2-5
Table 2.1.3-1	Status of TRU waste generation	2-8
Table 2.2.1-1	Assumed waste treatment and packaging (facility operation)	2-10
Table 2.2.1-2	Assumed waste treatment and packaging (facility dismantling)	2-11
Table 2.2.2-1	Pre-conditions for estimating waste volumes	2-14
Table 2.2.2-2	Estimated numbers of waste packages	2-15
Table 2.2.2-3	Estimated number of waste packages (private sector reprocessing operations)	2-16
Table 2.2.2-4	Estimated number of waste packages generated (private sector MOX operation)	2-17
Table 2.2.2-5	Estimated waste packages (private sector reprocessing, dismantling)	2-17
Table 2.2.2-6	Estimated waste packages (private sector MOX dismantling)	2-18
Table 2.2.2-7	Estimated waste packages (JAEA reprocessing operation)	2-19
Table 2.2.2-8	Estimated waste packages (JAEA MOX operation)	2-20
Table 2.2.2-9	Estimated waste packages (JAEA reprocessing, dismantling)	2-21
Table 2.2.2-10	Estimated waste packages (JAEA MOX dismantling)	2-22
Table 2.2.2-11	Estimated waste packages (returned waste)	2-22
Table 2.3-1	Pre-conditions for determining concentrations of radioactive materials	2-24
Table 2.3-2	Variation in fission product content of reprocessing waste with the degree of burn-up of spent fuel	2-25
Table 2.3-3	Activity of radioactive materials in each waste package (Bq Mg ⁻¹) (private sector)	2-26
Table 2.3-4	Activity of radioactive materials in each waste package (Bq Mg ⁻¹) (returned waste)	2-26
Table 2.3-5	Activity of radioactive materials in each waste package (Bq Mg ⁻¹) (JAEA)	2-27
Table 2.3-6	Activity of radioactive materials in each waste package (Bq) (private sector)	2-28
Table 2.3-7	Activity of radioactive materials in each waste package (Bq) (returned waste)	2-28
Table 2.3-8	Activity of radioactive materials in each waste package (Bq) (JAEA)	2-29
Table 2.3-9	Comparison of quantities of major nuclides in different radioactive materials, per unit hulls and ends	2-30

Table 2.4.1-1	Relative important nuclides	2-31
Table 2.4.1-2	Activities used to decide form of waste disposal (Bq Mg ⁻¹)	2-32
Table 2.4.2-1	Volume of waste packages in each disposal category	2-33
Table 2.4.2-2	Activities of major nuclides in each disposal category	2-34
Table 2.5.1-1	Quantities of NaNO ₃ in the various waste packages	2-35
Table 2.5.1-2	Quantities of organic material in the various waste packages	2-36
Table 2.5.1-3	Contents of each waste package (private sector)	2-37
Table 2.5.1-4	Contents of each waste package (returned waste)	2-37
Table 2.5.1-5	Contents of each waste package (JAEA)	2-38
Table 2.5.2-1	Calorific power of various waste packages	2-39
Table 2.5.3-1	Waste packages sorted by immobilization media	2-41
Table 2.5.3-2	Sorting of waste packages by form	2-41
Table 2.5.3-3	Sorting of waste packages by content	2-42
Table 2.5.3-4	Sorting of waste packages by heat production rate	2-42
Table 2.5.3-5	Sorting by C-14 and I-129 content	2-43
Table 2.5.4-1	Sorting of wastes	2-44
Table 2.5.4-2	Breakdown of sorting for waste packages	2-45
Table 3.1-1	Requirements considered in the design of the disposal facility	3-1
Table 3.1.1.1-1	Summary of the main functions of each engineered barrier component	3-5
Table 3.2.1.1-1	Waste package concepts	3-8
Table 3.2.1.2-1	Required functions of the buffer material	3-9
Table 3.2.1.2-2	Parameter settings used	3-13
Table 3.2.1.2-3	Example specifications of buffer material	3-17
Table 3.2.1.3-1	Required functions of the filler	3-18
Table 3.2.2.1-1	Mechanical property values used in the analyses	3-21
Table 3.2.2.1-2	Cases used in analysis	3-22
Table 3.2.2.1-3	Summary of analytical result	3-23
Table 3.2.2.1-4	Parameters used in the MBC analysis	3-25
Table 3.2.2.1-5	Cases used in analysis	3-25
Table 3.2.2.1-6	Summary of analytical results	3-26
Table 3.2.2.1-7	List of analytical cases	3-28
Table 3.2.2.1-8	Summary of results	3-29
Table 3.2.2.1-9	Configuration and scale of tunnels in each rock type	3-29
Table 3.2.2.2-1	Desired properties of the structural framework	3-30
Table 3.2.2.2-2	Thermophysical properties	3-32
Table 3.2.2.2-3	Cases used in the thermal analysis	3-34
Table 3.2.2.2-4	Thermal analysis result for the reference case	3-36
Table 3.2.2.2-5	Example results of the thermal analysis	3-37

Table 3.2.2.2-6	Summary of spacing between disposal tunnels	3-39
Table 3.2.2.3-1	Candidate backfill materials for the disposal facility	3-48
Table 3.2.2.3-2	Example specifications for bentonite backfill material	3-48
Table 3.2.2.3-3	Example specifications of the hydraulic plug	3-49
Table 3.3-1	Modelling of phenomena for analytical evaluation	3-54
Table 3.3-2	Analytical cases	3-56
Table 3.3-3	Model of the engineered barriers (for Group 1 waste in soft rock)	3-60
Table 3.3-4	Evolution of chemical properties up to final step	3-61
Table 3.4.2.1-1	Main equipment in facility for receiving and inspecting waste	3-75
Table 3.4.2.1-2	Processing and manufacturing techniques	3-76
Table 3.4.2.1-3	Management and manufacturing processes	3-77
Table 3.4.2.2-1	Emplacement techniques of buffer material	3-84
Table 3.4.2.2-2	In situ buffer emplacement techniques (1/4)	3-85
Table 3.4.2.2-3	In situ buffer emplacement techniques (2/4)	3-86
Table 3.4.2.2-4	In situ buffer emplacement techniques (3/4)	3-87
Table 3.4.2.2-5	In situ buffer emplacement techniques (4/4)	3-88
Table 3.4.2.2-6	Relationship between construction method of buffer material and attained density/hydraulic conductivity	3-90
Table 3.4.3.1-1	Operation concept for the TRU waste repository	3-101
Table 3.4.3.1-2	Basic concept for radiation control	3-103
Table 3.4.3.2-1	Number of handled waste units	3-105
Table 3.4.3.2-2	Material for transportation and weight	3-106
Table 3.4.3.2-3	Example of waste transportation	3-108
Table 3.4.3.2-4	Waste forms in circular disposal tunnel	3-109
Table 3.4.3.2-5	Transportation of waste to a disposal tunnel and required equipment (circular disposal tunnel)	3-110
Table 3.4.3.2-6	Example waste emplacement method (circular disposal tunnel)	3-111
Table 3.4.3.2-7	Waste forms in horseshoe-shaped disposal tunnel	3-112
Table 3.4.3.2-8	Example waste emplacement method (horseshoe-shaped disposal tunnel)	3-113
Table 3.4.3.2-9	Example of mortar filling method	3-116
Table 3.4.4.1-1	Backfill materials and location	3-120
Table 3.4.4.1-2	Backfilling methods with bentonite	3-121
Table 3.4.4.1-3	Appropriate backfilling methods for the main shaft and connecting tunnel	3-124
Table 3.4.5-1	Main measurements/information during stepwise construction of the repository (example of high-level waste disposal)	3-126
Table 4.1.1-1	Items for investigation corresponding to the safety evaluation system, and their content	4-4
Table 4.2.1-1	Requirements for a safety assessment	4-7

Table 4.2.2-1	Geological information used in assessments	4-8
Table 4.2.3-1	Classification of the waste packages in each waste package group	4-8
Table 4.2.3-2	Model inventory for each waste package type used in safety assessment (after 25 years of storage) (Bq)	4-10
Table 4.2.4-1	Parameters describing the features of the disposal facility for each waste package group	4-11
Table 4.3.1-1	Comprehensive FEP list for the geological disposal of TRU waste	4-15
Table 4.3.2-1	Integration of comprehensive FEPs for the purpose of classification	4-17
Table 4.3.3-1	FEPs that can potentially be excluded from the safety assessment	4-19
Table 4.3.3-2	Two arguments concerned with FEPs for which judgment is reserved	4-21
Table 4.3.6-1	Types of FEPs, summary FEPs, scenarios and repository conditions that are important for safety assessment	4-27
Table 4.4.2.2-1	Concrete and mortar components for evaluation	4-32
Table 4.4.2.2-2	Equilibrium constants for model solid solutions as functions of Ca/Si ratios	4-37
Table 4.4.2.2-3	Initial hydrates and secondary mineral considered in the chemical model of cementitious material	4-38
Table 4.4.2.2-4	Specifications for an analysis of variations in hydraulic conductivity in a disposal facility without bentonite	4-39
Table 4.4.2.2-5	Examples of the times for which specified chemical conditions occur, based on the analysis of alteration in a disposal facility without bentonite	4-41
Table 4.4.2.2-6	Specification of buffer material (based on the evaluation in Section 3.2.1.2)	4-46
Table 4.4.2.2-7	Mineralogical composition of bentonite	4-46
Table 4.4.2.2-8	Potential products of the interaction of bentonite with hyperalkaline fluids	4-47
Table 4.4.3-1	Minerals considered in each barrier material	4-68
Table 4.4.3-2	Physical properties and parameters values used in the analyses of alkali constituents	4-68
Table 4.4.4-1	Hydraulic conductivity specified for cement mortar	4-75
Table 4.4.4-2	Values specified for the hydraulic conductivity of the backfill	4-76
Table 4.4.4-3	Summary of analytical cases	4-78
Table 4.4.4-4	Darcy flow velocity in each region and groundwater flux in the region affected by drilling	4-79
Table 4.4.7-1	Effects of microbial activity	4-93
Table 4.4.7-2	Substrates and microbial activities in a geological repository	4-95
Table 4.4.8-1	Comparison of absorbed dose rate on waste surfaces	4-101
Table 4.4.8-2	Conditions specified for evaluating the behaviour of oxidizing agents in mortar and buffer material	4-108
Table 4.4.10.1-1	Results of gas migration analysis	4-121
Table 4.4.10.2-1	Maximum C-14 release rate and maximum generation rate in the case that all C-14 is	

	released as CH ₄	4-124
Table 4.4.10.4-1	Summary of results from evaluations of environmental conditions	4-129
Table 4.4.10.4-2	Summary results from an evaluation of discrete phenomena (1/2)	4-130
Table 4.4.10.4-2	Summary results from an evaluation of discrete phenomena (2/2)	4-131
Table 4.5.1-1	Relationship between scenarios, analytical cases and uncertainties	4-134
Table 4.5.2-1	Radionuclide ratio in metals of Group 2 waste	4-144
Table 4.5.2-2	Radionuclide dissolution period in Group 2 waste	4-144
Table 4.5.2-3	Solubility enhancement factors (SEF) for actinides and Tc	4-145
Table 4.5.2-4	Specified values for solubility enhancement factor (SEF) and sorption reduction factor (SRF)	4-145
Table 4.5.2-5	Solubility of specified elements	4-146
Table 4.5.2-6	Radionuclide effective diffusion coefficients for compacted bentonite (unit: m ² /s)	4-147
Table 4.5.2-7	Sorption distribution coefficients for elements in cement mortar (unit: m ³ /kg)	4-149
Table 4.5.2-8	Sorption distribution coefficient for elements in the buffer material (unit: m ³ /kg) ...	4-150
Table 4.5.2-9	Hydraulic conductivities used in the EBS and host rock	4-150
Table 4.5.2-10	Groundwater flow velocity in the engineered barriers and the groundwater flux in the excavation disturbed zone	4-151
Table 4.5.2-11	Sorption distribution coefficients of elements in the host rock (unit: m ³ /kg)	4-157
Table 4.5.2-12	Summary boundary conditions in the Reference Case model	4-158
Table 4.5.2-13	Dose conversion factors ((Sv/y)/(Bq/y))	4-160
Table 4.5.3-1	Summary of parameters in alternative cases for effects of alteration of the engineered barrier materials	4-171
Table 4.5.3-2	Summary of parameters used in alternative cases for host rock data uncertainty	4-172
Table 4.5.3-3	Summary of parameters used in alternative cases on effects of initial oxidizing conditions	4-173
Table 4.5.3-4	Summary of parameters used in alternative cases on effects of colloids	4-174
Table 4.5.3-5	Summary of parameters used in alternative cases on effects of organic material	4-174
Table 4.5.3-6	Summary of parameters used in alternative cases on gas influences	4-175
Table 4.5.3-7	Summary of parameters for alternative geological environment cases	4-176
Table 4.5.3-8	Alternative groundwater origin case	4-177
Table 4.5.3-9	Sensitivity analysis of parameters in natural barrier	4-178
Table 4.5.4-1	Parameter ranges and constant parameter values used in the comprehensive sensitivity analyses (for Group 1 waste)	4-193
Table 4.5.4-2	Example of the successful condition	4-198
Table 4.5.5-1	Assumptions made in conceptual model of uplift/erosion	4-204
Table 4.5.5-2	Transport pathway and specification of the biosphere in the uplift/erosion scenario	4-205

Table 4.5.5-3	Assumptions made in the conceptual model of climate/sea-level change	4-208
Table 4.5.5-4	Transport pathway and biosphere specifications in the climate/sea-level change scenario (inland zone)	4-210
Table 4.5.5-5	Transport pathway and biosphere specifications in the climate/sea-level change scenario (coastal zone)	4-211
Table 4.5.5-6	Dose conversion factors used in climate/sea-level change scenario	4-212
Table 4.5.5-7	Transport pathway and biosphere conceptualization in the sealing defect scenario	4-215
Table 4.5.5-8	Transport pathway and biosphere specifications in the deep well scenario	4-217
Table 4.5.5-9	Dose conversion factors for river water/plain model and deep well/plain model	4-218
Table 4.5.5-10	Transport pathway and biosphere specifications for drilling near the disposal facility	4-220
Table 4.5.6-1	Evaluation target for intrusion by drilling scenario	4-227
Table 4.5.6-2	Parameter settings for calculating external dose to drill core observers and drilling operators	4-229
Table 4.5.6-3	Parameter settings for calculating radiation doses to drill core observers	4-229
Table 4.5.6-4	Parameter settings for calculating external dose to drilling operators	4-230
Table 4.5.6-5	Projected cross-sectional area in the risk calculation for the disposal tunnels for each waste group	4-231
Table 5.3.2-1	Newly specified assessment parameters	5-5
Table 5.3.2-2	Revised assessment parameters	5-6
Table 5.3.4-1	Revised assessment parameters	5-7
Table 5.4.1-1	Results of dose assessments for concrete vault disposal	5-8
Table 5.4.2-1	Results of dose assessments for intermediate-depth disposal	5-10
Table 6.2.1-1	Disposal concepts for radioactive waste in other countries	6-4
Table 6.2.2.2-1	Thermal properties of the engineered barriers and host rock	6-9
Table 6.2.2.3-1	Parameters and values used	6-14
Table 6.2.2.3-2	Relationships between increases in the concentrations of major elements and ISA concentrations	6-19
Table 6.2.2.4-1	Parameters and values used	6-23
Table 6.2.2.4-2	Maximum nitrate concentrations attained at different points in the plume upstream from the TRU waste disposal facility during a period of 100,000 years	6-26
Table 6.2.2.4-3	Maximum nitrate concentrations attained at different points in the plume perpendicular to the flow direction during a period of 100,000 years	6-26
Table 6.2.2.4-4	Maximum nitrate concentrations of the plume attained at different points on the downstream side from TRU waste disposal facility over a period of 100,000 years	6-27
Table 6.2.2.5-1	Minerals which contribute to the geochemical reaction	6-33

Table 6.2.2.5-2	Chemical composition of cement (wt%)	6-34
Table 6.2.2.5-3	Specified mix of model mortar	6-34
Table 6.2.2.5-4	Mineral composition of cement (mol/dm ³ water)	6-34
Table 6.2.2.5-5	Minerals considered in each barrier material	6-35
Table 6.2.2.5-6	Parameter values, including physical properties, used in the simulations of a high-pH plume	6-35
Table 6.2.2.5-7	Alteration of bentonite in alkaline solution	6-45
Table 6.3.2-1	Estimated upper limits of nuclide concentrations in TRU waste	6-54
Table 6.3.4-1	Amounts of waste generated in each disposal category (upper limit of α -emitting nuclide concentration: 100 GBq/t)	6-58
Table 6.3.4-2	Comparison between amounts of waste generated in each disposal category	6-59
Table 6.3.4-3	Radioactivity in each disposal category (Bq)	6-60
Table 6.3.5-1	Result of dose assessment for intermediate-depth disposal (upper limit of α -emitting nuclide concentration: 100 GBq/t)	6-61
Table 6.3.5-2	Dose comparison for intermediate-depth disposal	6-62
Table 6.4.1.2-1	Assumed waste treatment and waste packages (returned waste)	6-67
Table 6.4.1.3-1	Estimated quantities of waste generated (returned waste including high-level waste)	6-68
Table 6.4.1.3-2	Comparison between estimated quantities of TRU waste generated	6-69
Table 6.4.1.4-1	Concentrations of radionuclides in each type of waste (returned waste) (1/2)	6-70
Table 6.4.1.4-1	Concentrations of radionuclides in each type of waste (returned waste) (2/2)	6-71
Table 6.4.1.4-2	Amount of nitrate in the waste (returned waste)	6-72
Table 6.4.1.4-3	Amount of organic material in the waste (returned waste)	6-72
Table 6.4.1.4-4	Heat generation rate of waste (returned waste)	6-73
Table 6.4.1.5-1	Estimated amounts of TRU waste generated in each disposal category (upper limit of α -emitting nuclide concentration for intermediate-depth disposal: 1 GBq/t)	6-74
Table 6.4.1.5-2	Estimated quantities of TRU waste generated in each disposal category (upper limit of α -emitting nuclide concentration for intermediate-depth disposal: 100 GBq/t)	6-75
Table 6.4.1.5-3	Amount of radioactive substances in each disposal classification (Bq)	6-76
Table 7.2-1	Iodine immobilization techniques and performance assessment models	7-5
Table 7.2-2	Structure and chemical composition of iodine immobilization solids	7-6
Table 7.3-1	Characteristics of different types of packages for the long-term containment of C-14	7-11
Table 7.3-2	Metal packages for containment of C-14	7-12
Table 7.4-1	Solidification time of HFSC mortar and compressive strength	7-15
Table 8.3-1	Fundamental research and development (waste database)	8-8
Table 8.3-2	Fundamental research and development (disposal technology)	8-9
Table 8.3-3	Fundamental research and development (performance assessment (1/2))	8-10

Table 8.3-3	Fundamental research and development (performance assessment (2/2))	8-12
Table 8.3-4	Fundamental research and development (alternative technologies)	8-13
Table 8.3-5	Development of implementation technologies (detailed design methods)	8-14
Table 8.3-6	Development of implementation technologies (construction technologies, etc.)	8-15

Chapter 1 Introduction

Nuclear fuel cycle policy in Japan calls for implementation of safe, reliable disposal measures for all categories of radioactive waste, including waste containing transuranic radionuclides (TRU waste) and high-level waste (HLW).

Against this background, Japan's reprocessing activities and the operation of its MOX fuel fabrication facilities, both of which produce TRU waste, will be reviewed in this report, taking into account results from relevant studies, technical data on treatment facilities for other types of radioactive waste and the state of development of the required institutional framework. Appropriate consideration is also given to the status of safety assessments and technological developments in foreign radioactive waste management programs.

This technical review of the disposal of TRU waste has been prepared as a collaborative effort of the electric power utilities, the Japan Nuclear Cycle Development Institute (JNC; since October 2005 the Japan Atomic Energy Agency) and other involved organizations. It updates the report entitled "Progress Report on the Disposal Concept for TRU Waste in Japan", which was published in 2000, and reflects the latest results of the research carried out by the utilities and JNC since the publication of this earlier report. The aim is to establish and demonstrate state-of-the-art technologies for the safe disposal of TRU waste.

1.1 Context of the report

1.1.1 Status of studies on TRU waste disposal in Japan

In March 2000, the electric power utilities and JNC published a summary document entitled "Progress Report on the Disposal Concept for TRU Waste in Japan" (1st TRU progress report; TRU Coordination Team, 2000). The report addressed the disposal of TRU waste arising from fuel reprocessing, MOX fuel fabrication and dismantling of nuclear facilities and the possibilities for disposing of this waste under representative geological conditions in Japan. It also highlighted technological issues to be addressed in future research and development programs.

Also in March 2000, the Advisory Committee on Nuclear Fuel Cycle Backend Policy of the Atomic Energy Commission of Japan referred to the 1st TRU progress report and studies on waste disposal methods carried out in Japan up to that time in preparing its summary report "Basic Concept for the Disposal of Radioactive Waste Containing Transuranic Radionuclides" (Advisory Committee Report; AEC, 2000a). The report discussed how, using criteria such as types and concentrations of radionuclides, this waste can be divided into three categories: that destined for disposal in shallow concrete vaults (shallow disposal), that for disposal below generally accessible depths (50-100 m; intermediate-depth disposal) and that suitable for deep geological disposal. The report also highlighted issues to be considered in future technical developments, as follows:

- In order to rationalize and refine disposal facility designs and build confidence in safety assessments, the following issues must be considered:
 - acquisition of relevant test data;
 - accurate understanding of key phenomena (for example alteration of cement, reactions between alkaline solutions/buffer material and the rock and the effects of nitrate and gas generation) and construction of assessment models taking these phenomena into account.

- Further research into improving the ability of the safety barriers to retain iodine.

- Development of a waste package database and quality management procedures and validation methods that will build confidence in the quality of waste package production.

The publication of the Advisory Committee Report was followed by a report of the Atomic Energy Commission entitled “On the Treatment and Disposal of Transuranic Radionuclide Bearing-Radioactive Waste” (AEC, 2000b). This states that the responsible authorities should take into consideration the findings in the Advisory Committee Report and, against this background, should cooperate in moving actively towards implementing disposal. The report also states that it is necessary to consider basic concepts for safety regulations and to establish safety standards and relevant legislation. It also highlights the importance of providing accurate and easily understandable information, while at the same time understanding the need to obtain feedback from concerned stakeholders, which is an important aspect of policy-making. The “Long-term Program for Development and Utilization of Nuclear Energy” (AEC, 2000c), which was revised in November 2000, presents issues that should be addressed by the waste producers in progressing towards implementation of radioactive waste disposal, including TRU waste. These issues are described below.

“In the case of radioactive waste for which concrete disposal measures have not yet been decided, it is necessary, in order to ensure safe and efficient management from the beginning, for the waste producers to cooperate closely in developing proposed strategies.

Besides high-level waste, there are other types of waste that require geological disposal. It is important that studies on disposal of these other wastes should make appropriate use of results from research on high-level waste disposal. It is also important to develop waste treatment and disposal technologies taking into account the diverse characteristics of these wastes.”

Under the auspices of the Advisory Committee on Energy and Natural Resources, between October 2003 and January 2004 the cost structure for the entire back-end project was analyzed by a subcommittee of the electricity utilities, in preparation for the start of commercial operation of a reprocessing plant by Japan

Nuclear Fuel Limited (JNFL) (ANRE, 2004). The expected costs of TRU waste disposal were presented, based on estimates of the cost of a high-level waste repository.

The Atomic Energy Commission of Japan has developed fundamental concepts for disposal of all types of radioactive waste produced by the nuclear power plants. Safety principles that apply to disposal of all waste types have also been formulated by the International Atomic Energy Agency (IAEA, 1995). Based on these documents, a committee on atomic energy safety was set up to consider key regulatory and safety issues for all wastes ranging from low-level to high-level. The results of the deliberations are summarized in a report entitled “Common Important Issues for Safety Regulations of Radioactive Waste” (NSC, 2004).

The electricity utilities, JNC and responsible government agencies have pursued research and development that targets the key technical issues described in the 1st TRU progress report and the Advisory Committee Report. In particular, in their capacity as TRU waste producers, the utilities and JNC (now JAEA) will continue to cooperate on research and development aimed at refining and rationalizing the disposal concept, improving understanding of the particular phenomena associated with TRU waste disposal and enhancing confidence in safety assessment methods.

Table 1.1-1 presents the status of discussions and the development of safety regulations and standards for the various categories of radioactive waste.

Table 1.1-1 Status of discussions, safety regulations and safety standards
for radioactive waste disposal (as of March 2005)

Waste classification		Atomic Energy Commission of Japan	Nuclear Safety Commission of Japan			Applicable laws and safety regulations		
		Basic disposal concept	Concept for safety regulation	Upper limit of concentration	Guidelines for safety review	Government decree	Regulation	Notification of technical details
High-level waste		Already evaluated (May 1998)	Already evaluated (November 2000)	-	Later consideration	To be formulated in future		
Low-level radioactive waste	Waste generated by nuclear facilities							
	Low-level waste with concentrations above those decreed by government	Already evaluated (October 1998)	Already evaluated (September 2000)	Already evaluated (September 2000)	Under consideration	Already established (December 2000)	To be formulated in future	To be formulated in future
	Homogeneous solid waste, miscellaneous solid waste, etc.	Already evaluated (August 1984)	Already evaluated (October 1985)	Already evaluated (February 1987, June 1992)	Already evaluated (March 1988)	Already established (March 1987, September 1992)	Already established (January 1988, September 1993)	Already established (January 1988, September 1993)
	Very low-level radioactive waste (VLLW) (concrete waste, etc.)			Already evaluated (June 1992)	Already evaluated (January 1993)	Already established (September 1992)	Already established (February 1993)	
	As above (VLLW) (metal etc. waste)			Already evaluated (September 2000)	Under consideration	Already established (December 2000)	Improvement in future	
	Radioactive waste including TRU			Already considered (April 2000)	Under consideration (June 2000~)	Later consideration	Later consideration	To be formulated in future
	Uranium waste	Already evaluated (December 2000)	Later consideration	Later consideration	Later consideration	To be formulated in future		
RI / laboratory waste	Already evaluated (June 1998)	Already evaluated (RI waste: January 2004) Under consideration (laboratory waste)	Later consideration	Later consideration	To be formulated in future			
No need to handle as radioactive waste	Clearance level	Already evaluated (August 1984)	Already evaluated (nuclear reactors: July 2001, facilities using nuclear fuel: April 2003) Under consideration (facility for RI) Later consideration (radioactive waste including TRU, uranium waste)			To be formulated in future		
	Clearance level validation		Already evaluated (nuclear reactors: July 2001 onwards) Later consideration (others)					

1.1.2 Objective of this report

In other countries with radioactive waste disposal programs, a stepwise or phased approach is adopted for research and development and for implementing disposal (e.g. OECD/NEA, 2004). At each stage, discussions are held and decisions made to provide appropriate input for the next step. At the same time, research institutions and other organizations provide input on key aspects of the long-term safety of disposal, to allow this to be reflected in the discussions and decision-making process. The current status of technological developments and technical issues remaining open are documented in integrated reports.

The following are examples of technical reports produced in stepwise investigation programs: the U.S. Department of Energy's EIS report for the Yucca Mountain Project (DOE, 2002); the Project Gewähr report series (Nagra, 1985), the Kristallin-1 report (Nagra, 1994) and the Entsorgungsnachweis report (Nagra, 2002) in Switzerland; the SR-97 report of SKB, Sweden (SKB, 1999) and the "Dossier 2001" of ANDRA, France (ANDRA, 2001). In Japan, the 1st progress report H3 (PNC, 1992) and the 2nd progress report H12 (JNC, 2000) on geological disposal of high-level radioactive waste, the 1st TRU progress report and the present report (the 2nd TRU progress report) have all been produced in the context of a staged program.

Production of TRU waste from the operation of JNFL's commercial reprocessing plant will commence in the near future. In addition, the new organization established in October 2005 by the merger of the Japan Nuclear Cycle Development Institute and the Japan Atomic Energy Research Institute (Japan Atomic Energy Agency, JAEA) will be responsible for implementing environmentally acceptable disposal measures for TRU waste. The time is therefore right to evaluate the procedures for implementation, disposal and site selection and to move forward with establishment of safety standards and the institutional framework for regulating safety.

Against this background, the utilities and JNC prepared the 2nd TRU progress report. Based on the latest information on waste packages and projected waste characteristics, the report presents a refined geological disposal concept for TRU waste, with the aim of improving confidence in the safety of disposal in a wide range of geological environments in Japan. To allow integration of the latest results of relevant research, the cooperation of the Central Research Institute of the Electric Power Industry (CRIEPI) and the Radioactive Waste Management Funding and Research Center (RWMC) was obtained.

A drafting team was set up to bring together all the relevant work and to prepare the final report. Three sub-working groups (SWG) were also established: a database SWG to prepare a database of waste characteristics and nuclide migration data, a disposal technology SWG for developing and refining the disposal facility design and a performance assessment SWG for evaluating the safety of the engineered and natural barriers. The report was prepared through active cooperation between these groups (cf. Appendix 1A).

1.1.3 Approach to preparing the 2nd TRU progress report

In preparing the 2nd TRU progress report, the technology development tasks identified in the Advisory Committee Report and the 1st TRU progress report were addressed. Information on disposal concepts and the safety of each potential disposal system - deep geological disposal, shallow disposal and intermediate-depth disposal - was based on the updated waste database, which reflects the results of the latest research and development and state-of-the-art waste treatment methods. In the 2nd TRU progress report, each disposal concept is evaluated by considering possibilities for improving repository operation and safety regulations. Activities such as site selection are not evaluated and a review of safety regulations and standards is presently being carried out. These activities can be regarded as part of the generic research and development stage and take into account a wide range of geological conditions.

The 2nd TRU progress report takes into account the data, mass transport models and safety assessment scenarios that were applied for a wide range of geological environments in the H12 report on HLW disposal, and which are based on the disposal concept described in the 1st TRU progress report. The details of the disposal concept are presented, together with a comprehensive safety assessment and supporting arguments.

In view of concepts being considered in other countries, an evaluation of co-disposal of high-level waste at the same site as TRU waste highlights approaches for preventing mutual interference between the two disposal facilities. The evaluation concludes that, for established disposal methods, mutual adverse influences can be ruled out.

A report entitled "Reference Radionuclide Concentration Values for Low-level Radioactive Solid Waste to be Land Disposed" (NSC, 1986) considered a shallow (concrete vault) disposal system for low-level radioactive waste generated by nuclear power plants; applicable laws and safety regulations have already been developed and implemented. JNFL is presently carrying out shallow disposal of low-level waste at Rokkasho village in Aomori Prefecture.

In the 2nd TRU progress report, waste containing alpha nuclides destined for concrete vault (shallow) disposal is distinguished using its upper radionuclide concentration limit, in accordance with the law on regulation of nuclear fuel/nuclear fuel material and nuclear reactors. A review was carried out, based on the disposal concept and safety assessment presented in the 1st TRU progress report.

For the intermediate-depth disposal system, a report on disposal of low-level waste generated by nuclear reactors with upper concentration limits that exceed current legal limits (AEC, 1998) considers the concept for a disposal facility and for ensuring safety during and after the management period for this waste. The law on the regulation of nuclear fuel/nuclear fuel material and nuclear reactors mentioned above has been

revised based on a document entitled “Reference Radionuclide Concentration Values for Low-level Radioactive Solid Waste to be Land Disposed (3rd Interim Report)” (NSC, 2000a). The upper concentration limit for waste generated from nuclear reactors was also based on the safety standards in this report. Additionally, a report on safety regulation of radioactive waste disposal discussed the application of risk in safety assessments that consider uncertainty in long-term repository performance. The application of risk to both the potential for a scenario to occur and the effects of the scenario was considered. The report stressed the importance of considering Japan’s regulatory system promptly, taking into account the introduction of risk as a basis for establishing regulations in most other countries. Hence, the requirement to ensure safety and the corresponding safety assessment methodologies for intermediate-depth disposal of TRU waste will be evaluated in depth in the future. Bearing this in mind, the 2nd TRU progress report presents an evaluation for a groundwater scenario based on parameter ranges consistent with the nuclide partition coefficient (K_d) values for cementitious material used in the evaluation of the geological disposal system and the disposal concept and safety assessment presented in the 1st TRU progress report.

In the Advisory Committee Report, wastes for intermediate-depth and geological disposal are classified as those with an average alpha concentration provisionally specified to be in the order of several GBq/t (maximum concentration of several tens of GBq/t). The 2nd TRU progress report presents a re-evaluation of the alpha concentrations used for classification, to ensure that they are reasonable in terms of their implications for safety.

Additional analyses aimed at rationalization were carried out, taking into account the large reduction in waste volumes, storage requirements and required transportation that will be possible with modification of the methods for returning TRU waste from overseas.

The actual circumstances of geological disposal of TRU, including the institutional framework, the disposal site and the specific geological environment, have not yet been decided. Consequently, work was aimed at improving safety margins and flexibility in selecting a geological environment. This was done by improving existing waste technology and evaluating alternative technologies and addressing processes involving large uncertainties, including the effects of nitric acid and bentonite-alkaline solution reactions.

To ensure transparency and traceability in preparing the 2nd TRU progress report, consideration was given to making the reasoning behind scientific judgments and decisions more clear. Meetings to exchange information were held with domestic experts (June 2004 and April 2005) and review meetings were held with Nagra (December 2003, April 2004 and February 2005). At these meetings, the parameters and data used in the technical assessments and the assessment models were discussed. Many comments were received and are reflected in the final version of the report. Additionally, to facilitate exchange of opinions with experts from various countries and collection of information on current progress in research and development, the 3rd International TRU Workshop was held in Oxford, UK, in January 2005. The results

of the meeting are also taken into account in this report.

The justification for selecting the data and parameter values used in the analyses in the report, together with supporting detailed analyses, have been published separately as a collection of supporting material (CD-ROM, <http://www.fepc-atomic.jp/nuclear/waste/tru/002.html>). These online documents (in Japanese) include a waste database, a nuclide migration database, design evaluation and analysis documentation, safety assessment scenarios, discrete phenomena analyses, safety assessments and details of the evaluation of co-site disposal.

1.2 Safety assurance for a geological repository

1.2.1 Background

In the safety assessment of a geological repository for radioactive waste, it is important to show that long-term safety can be assured after closure of the facility. In this section, safety assurance concepts for a geological repository for TRU waste are considered with reference to international trends and existing Japanese assessments.

The International Atomic Energy Agency (IAEA) has published a document entitled “Geological Disposal of Radioactive Waste: Draft safety requirements: Safety Standards Series DS154” (DS154; IAEA, 2004), which was produced with the aim of safeguarding geological repositories and as a basis for defining safety standards. It presents the basic requirements that must be fulfilled to ensure the safety of disposal in accordance with existing principles.

DS154 presents the requirements for ensuring long-term radiological safety, particularly after closure of the disposal facility. It is stated in Requirement 2 that it is the responsibility of the regulatory agency to “establish the regulatory requirements for the development of geological disposal facilities”. Requirement 3 states that “The operator of a geological disposal facility shall be responsible for its safety”. Additionally, it is the responsibility of the operator to “carry out safety assessments and develop a safety case, and carry out all the activities needed for siting, design, construction, operation and closure, in compliance with the regulatory requirements and within the national legal framework.”

In Japan, safety regulations corresponding to these requirements have been considered for a geological repository for high-level waste (NSC, 2000b), for a shallow disposal system for low-level waste from nuclear power plants (NSC, 1985, 1986, 1992, 2000a) and for a shallow disposal system for radioisotope (RI) waste (NSC, 2003). Also, common requirements for those wastes are documented in the “Common Important Issues for Safety Regulations of Radioactive Waste” (NSC, 2004). However, the principles for ensuring the safety of a geological repository for TRU waste have not yet been clearly defined. Although these principles will be considered in the future, it is important, before these deliberations, to obtain fundamental information and to perform related safety assessments. The results of these assessments will contribute to the development of necessary institutional framework and the planning of a geological repository for TRU waste.

In this report, Requirement 3 of DS154 is applied with appropriately, since the present report considers the framework for ensuring safety from the point of view of the waste producers. The evaluation of TRU waste disposal was carried out in accordance with this requirement. The safety requirements considered in this report are described in Section 1.2.2.

1.2.2 Safety requirements

As mentioned previously, the principles for ensuring the safety of a geological repository for TRU waste in Japan are not clearly defined at present. Therefore, the principles applying to the safety regulation of high-level waste disposal, examples from foreign countries and international agreements on disposal safety were evaluated. In the present report, the principles for ensuring the safety of the TRU repository are based on DS-154 which is consistent with the international principles on radioactive waste management (IAEA, 1995).

Not all the safety requirements of DS154 are considered to be important in the present technical evaluation of a TRU repository. In the present report, they are treated according to the following classification:

- (i) Not a requirement for waste producers and hence not applicable
- (ii) Not a requirement in the current technical evaluation and hence not applicable
- (iii) Applied appropriately as a prerequisite to the technical evaluation
- (iv) Applied appropriately as a common requirement of the technical evaluation
- (v) A requirement for understanding the volume and characteristics of the generated waste
- (vi) Applied appropriately as a disposal system design requirement
- (vii) Applied appropriately as a safety assessment requirement

The evaluation in the present report focuses on requirements (iii) to (vii). Of these, (iv) to (vii) of the technical requirements are considered as safety requirements in this report. Table 1.2-1 shows safety factors that are considered to be important in this report and the corresponding requirement number of DS154.

1.2.3 Safety standards

Different countries adopt different types of safety standards for geological disposal of radioactive waste. Some apply dose limits and analogous dose criteria, while others use risk criteria (e.g. NRC, 1982/2002; DSIN, 1991; BMU, 2001; RSA93, 1997; SSI, 1998/2000; STUK, 2001/2003; HSK & KSA, 1993).

In Japan, regulatory guidelines have been specified to ensure radiation protection after the management period for low-level waste suitable for shallow disposal (NSC, 1988). According to these guidelines, the dose should not exceed 10 $\mu\text{Sv}/\text{y}$ due to generally occurring phenomena and should not significantly exceed 10 $\mu\text{Sv}/\text{y}$ due to phenomena with a low likelihood of occurrence. The same guidelines are adopted for regulating the safety of the component of TRU waste destined for shallow disposal. However, although the same basic safety principles are being considered for intermediate-depth disposal and deep geological disposal (AEC, 1998; NSC, 2000b), concrete safety standards have not yet been developed.

In Japan, as in other countries, different disposal methods are being considered depending on the different

types and characteristics of radioactive waste. The “Common Important Issues for Safety Regulations of Radioactive Waste” (NSC, 2004) mentioned above points out that the detailed regulatory requirements should conform to the various methods of waste disposal and that separate approaches should be implemented for different wastes. However, it was also stated that the following basic considerations are relevant to safety regulation for every disposal method: (i) Introduction of a risk concept by taking into account the possibility that a particular scenario may occur; (ii) radiation protection standards taking into account the possibility that a particular scenario may occur; (iii) definition of the evaluation period, (iv) definition of the extent of treatment of human intrusion. It was also pointed out that the recommendations of the International Commission on Radiological Protection and regulatory trends in foreign countries should be taken into account.

In view of the above progress in Japan on definition of safety standards, the present report considers that, for TRU waste, an appropriate safety standard for a shallow disposal system is a dose rate of 10 $\mu\text{Sv}/\text{y}$ and that an appropriate standard for long-term safety after closure of facilities for intermediate-depth disposal and deep geological disposal, according to point (ii), is the same as that proposed in foreign countries (i.e. a dose rate 100 – 300 $\mu\text{Sv}/\text{y}$ or a risk of 10^{-5} – $10^{-6}/\text{y}$).

Table 1.2-1 Safety requirements in this report

General item	Requirement	Requirement no. in DS154 (IAEA, 2004)
(i) Common requirements for technical evaluation	-Specific demonstration of disposal facility safety and level of reliability of safety	Requirement 12
	-Consideration of temporal and spatial variations	Requirement 12
	-Showing that there is no problem that might compromise safety	Requirement 12
	-Presentation of unsolved problems	Requirement 12
	-Ensuring adequate documentation (justification, tracking performance, clarity)	Requirement 13
	-Exchanging opinions with internal and external reviewers	Requirement 13
	-Quality management (including models used, data and verification and validation of codes)	Requirement 23
(ii) Requirements for understanding the volumes and characteristics of generated waste	-Understanding the inventory and characteristics of radionuclides and related factors (thermal output, matrix, package, etc.)	Requirement 19
(iii) Disposal system design requirements	-Design of passive engineered barrier system with multiple safety function	Requirement 7
	-Design of disposal system for containment, low flow rate, diffusion, dispersion, dilution and retardation	Requirement 7 Requirement 8
	-Design of appropriate infrastructure and layout	Requirement 9, Requirement 15
	-Design to reduce hazardous phenomena and perturbations	Requirement 15
	-Design taking into account the interactions between different barrier components, with the aim of maximizing their complementary effects	Requirement 7, Requirement 15
	-Design to minimize the effects of human intrusion	Requirement 9
	-Design to ensure safety during operation	Requirement 12 Requirement 15
(iv) Safety assessment requirements	-Ensuring that the results of the safety assessment are sufficiently reliable	Requirement 6
	-Understanding the characteristics and processes that contribute to safety; identification and understanding of phenomena and processes that might be detrimental to safety	Requirement 6
	-Considering of uncertainty in the safety assessment	Requirement 6
	-Checking adequacy of research scope (assessment timescale, assessed events/scenarios, analytical case)	Requirement 12
	- Application of multiple arguments based on the results of sensitivity analysis, analysis of the significance of uncertainty, “what if” analysis, reserve FEPs, stylized approach, adopting complementary safety indicators and natural analogue research	Requirement 9 Requirement 12

1.3 Geological environment

The geological environment at the site for the disposal facility plays an important role in ensuring the long-term safety of waste disposal. Important functions of the geological environment include physically isolating the waste from the surface environment, preventing migration of nuclides from the repository and maintaining the long-term stability of the facility. When selecting a repository site, it is necessary to consider whether the geological environment will fulfill these roles adequately. The disposal of TRU waste is presently at the evaluation stage that precedes the site selection stage and, consequently, the actual geological environment of the repository cannot yet be described in detail. The H12 report (JNC, 2000) showed that, in Japan, there are geological environments that would offer long-term stability of the rock formations and hence the safety of a HLW disposal facility. The present report therefore assumes geological environment conditions based on this knowledge.

1.3.1 Principles for selecting the geological environment

The most appropriate disposal concept for TRU waste depends on the radioactivity of the waste. The waste is classified into that which is suitable for shallow burial and that which is suitable for other disposal methods. Analysis results have been presented for the latter (AEC, 2000a; TRU Coordination Team, 2000).

In the ongoing deep geological and shallow disposal projects in Japan, different research and development approaches are adopted and the geological environment conditions and parameter values must be selected differently in each case. In this report, the geological environments for deep geological disposal and shallow disposal were selected according to the principles outlined in the following sections.

1.3.1.1 Geological disposal

The concept of geological disposal is based on a multiple barrier system consisting of engineered and natural components. The geological disposal concept has to consider a wide range of geological environment conditions in Japan. At present, a generic geological environment is assumed for the repository project. The evaluation of the geological conditions in the 1st TRU progress report (TRU Coordination Team, 2000) was performed by considering the investigation results for the geological environment presented in the H12 report. A range of conditions was taken into consideration by establishing several typical patterns for the geological environment.

1.3.1.2 Concrete vault disposal and intermediate-depth disposal

Some TRU waste can potentially be disposed of by shallow burial. A general safety assessment of this disposal option for power plant operational and decommissioning waste has already been

performed (NSC, 1987; AEC, 1998; NSC, 2000a). Hence, in the work reported here, the geological conditions for this disposal concept were selected with reference to the evaluation of a generic geological environment by the Nuclear Safety Commission.

1.3.2 Range of geological environments and the reference case

1.3.2.1 Geological repository

This report sets the geological environment conditions based on the H12 report, which summarizes the large body of knowledge on the geological environment in Japan, and takes into account a range of non-site-specific geological conditions. The specification of the main parameters is described below and is summarized in Table 1.3-1.

Table 1.3-1 Specification of parameters for the geological environment for a repository

	A			B	C	D
Geography	Inland			<u>Inland</u>	Coast	Coast
Topographic features	Plain (hills, mountains)			<u>Plain</u> (hills, mountains)	Plain	Plain
Lithology	Sedimentary rock			<u>Crystalline rock</u>	Sedimentary rock	Crystalline rock
Groundwater origin	Precipitation			<u>Precipitation</u>	Oceanic	Oceanic
Transmissivity* ¹ (m ² /s)	10 ⁻¹⁰ (10 ⁻⁹ , 10 ⁻¹¹)			<u>10⁻¹⁰</u> (10 ⁻⁹ , 10 ⁻¹¹)	10 ⁻¹⁰ (10 ⁻⁹ , 10 ⁻¹¹)	10 ⁻¹⁰ (10 ⁻⁹ , 10 ⁻¹¹)
Hydraulic conductivity (m s ⁻¹)	10 ⁻⁹ (10 ⁻⁸ , 10 ⁻¹⁰)			/	10 ⁻⁹ (10 ⁻⁸ , 10 ⁻¹⁰)	/
Hydraulic gradient	0.01 (0.05)			<u>0.01</u> (0.05)	0.01	0.01
Host rock type* ²	SR-C	(SR-B)	(SR-D)	<u>HR</u>	SR-C	HR
Effective porosity (-)	0.3	0.2	0.45	<u>0.02</u>	0.3	0.02
Uniaxial compressive strength (MPa)	15	20	10	<u>115</u>	15	115
Disposal depth (m)	500			<u>1,000</u>	500	500
Biosphere	River water			<u>River water</u>	Seawater	Seawater

Note 1: Underlined entries indicate the reference conditions.

Note 2: In this report, ranges for permeability and strength, etc. of the host rocks are set separately for each evaluation, considering its purpose (e.g. whether aimed at tunnel stability analysis, nuclide migration analysis, etc.).

*1: The given values means the log average of the distribution of transmissivity

*2: HR (hard rock) dataset: 0.02 effective porosity and 115 MPa uniaxial compressive strength;
 SR-B (soft rock B dataset): 0.2 effective porosity and 20 MPa uniaxial compressive strength;
 SR-C (soft rock C dataset): 0.3 effective porosity and 15 MPa uniaxial compressive strength;
 SR-D (soft rock D dataset): 0.45 effective porosity and 10 MPa uniaxial compressive strength.

(1) Topography

Combinations of the features below were used to represent the topography, based on the overall topography of the Japanese archipelago. With the aim of representing the relationship between geography and water quality, two typical groups of land forms were selected for evaluation from these combinations of features: one inland with a plain and freshwater type groundwater and one coastal with a plain and saline groundwater. However, for a plain with freshwater type groundwater, several variations of topography (mountains and hills) were considered.

- Geography: inland, coast
- Topography: mountains, hills, plain (plateau and lowland)
- Water type: freshwater, seawater

(2) Rock type

In the H12 report, rock types in Japan are classified into six categories as follows:

Crystalline “acidic rock” (igneous rock, metamorphic rock)

Crystalline “basic rock” (igneous rock, metamorphic rock)

Pre-Neogene sedimentary rock (sandstone)

Pre-Neogene sedimentary rock (mudstone/tuff)

Neogene sedimentary rock (sandstone)

Neogene sedimentary rock (mudstone/tuff)

In this report, crystalline acidic rock (granite) and Neogene sedimentary rock (mudstone/tuff) were selected as typical rocks, defined by their mechanical properties and taking into account their influence on nuclide migration and the mechanical stability of caverns. Each of these two lithologies was assumed to occur individually in combination with each of the two typical land form outlined above. Thus, in total, four typical cases were assumed.

(3) Geological structure

Nuclide migration from the repository through the natural barrier to the biosphere is considered to occur via long and complicated pathways because of the presence of geological structures, including faults. However, in the work reported here it was difficult to specify the nuclide migration pathways because a specific site has not been selected. The safety assessment in the H12 report considered the effect on the stability and performance of the repository of an active fault and an associated large-scale, high-permeability fracture zone located at a distance of 100 m.

Hence, in this report the reference case for the assessment assumes the thickness of the natural

barrier is 100 m . Additionally, in order to evaluate natural barrier performance, crystalline rock is treated as a fractured medium. In contrast, in cases where the natural barrier is a sedimentary-type basement rock, it was generally approximated as a porous medium or as a combination of fractured and porous rock. In this report, radionuclide migration in sedimentary-type host rock was evaluated using overlapping 1D parallel surface models, as described in the H12 report for the evaluation of fractured media in crystalline rock.

(4) Hydraulic conditions

The hydraulic gradient was based on a synthesis of information on groundwater levels in the H12 report. Values for the hydraulic gradient were then selected to correspond to each type of topography, as given below:

Plain (plateau and lowland): 0.01

Mountains, hills: 0.05

The reference migration analysis for fractured media also used the transmissivity data from the H12 report. The reported values for transmissivity distribution (log average value -9.99, logarithmic standard deviation 1.07) were obtained from a permeability test performed in a borehole in the Kamaishi mine. However, in order to evaluate the variation range of environmental transmissivity distribution, two alternative cases were considered, one with a transmissivity distribution one order of magnitude smaller and a second with a distribution one order of magnitude higher. Based on the information from the H12 report, the parameter range for hydraulic conductivity of basement rock was assumed to be $10^{-8} \text{ m s}^{-1} - 10^{-10} \text{ m s}^{-1}$ for crystalline and sedimentary rock. The reference case was specified as having the hydraulic conductivity of 10^{-9} m s^{-1} .

(5) Groundwater chemistry

The characteristics of groundwater depend on the surrounding topography. In this report, freshwater type groundwater, which is the most common type in Japan according to statistical analyses using groundwater data from present references in the H12 report, is selected as reference groundwater type and seawater type groundwater will be considered as the variation of reference groundwater type.

(6) Rock mechanics

The strength of the host rock is an important factor which affects disposal facility design from the viewpoint of ensuring cavity stability. The existing data on physical properties of basement rock in Japan were presented in the H12 report and used to derive a dataset of basic physical properties of

basement rock. In the present report, the data for a hard rock system (HR: average value of acidic rock) in the H12 report are used for crystalline rock. To represent the wide range of rock strength distribution shown by sedimentary rocks, three different cases were taken from the H12 report. The soft SR-C basement rock was taken as the reference case and the cases SR-B and SR-D were taken as alternatives.

1.3.2.2 Concrete vault disposal and intermediate-depth disposal

The possibility of disposing of a component of TRU waste in concrete vaults at a depth of several meters below ground surface or in intermediate-depth disposal facilities (e.g. 50 - 100 m) have been considered previously (AEC, 2000a, TRU Coordination Team, 2000). Hence, in this report, taking into account the evaluation of the Nuclear Safety Commission, the geological environment conditions for concrete vault disposal and intermediate-depth disposal are as shown in Table 1.3-2.

1.3.3 Summary

Referring to the ranges of geological conditions documented in the H12 report for deep geological disposal, four typical cases were specified. Additionally, for shallow disposal, the geological conditions in this report were established by considering the results of an assessment of shallow disposal of power plant waste performed by the Atomic Energy Commission of Japan and the Nuclear Safety Commission of Japan.

Table 1.3-2 Specification of geological environment conditions
for shallow and intermediate-depth disposal

	Item	Subsidiary item	Set value	Remarks
Shallow disposal	Geological conditions	Nuclide migration pathway	500 m	The value near the center of the range 100–1,000 m, which is stated to be reasonable in IAEA-TECDOC-401.
		Hydraulic conditions	Hydraulic conductivity	10^{-6} m s^{-1}
	Porosity		0.3	
	Hydraulic gradient	0.01		
Intermediate-depth disposal	Geological conditions	Nuclide migration path length	500 m	Use as porous media On-site: 200 m Off-site: 300 m
		Hydraulic conditions	Hydraulic conductivity	10^{-7} m s^{-1}
	Porosity		0.2	The porosity is set by considering the value in “Groundwater Handbook” (Chikasui Handobukku Henshu-iinkai, 1979) and the H12 report.
	Hydraulic gradient		0.01	The gradient is set based on the hydraulic gradient distribution quoted for lowland in the H12 report.

References

- AEC (1998): On the Disposal of Low-Level Radioactive Waste that Exceeds the Present Legislative Upper Concentration, Atomic Energy Commission of Japan, October 16th, 1998. [written in Japanese]
- AEC (2000a): Fundamental Considerations on the Treatment and Disposal of Transuranic Nuclide Bearing-Radioactive Waste, Special Committee on Backend Policy, Atomic Energy Commission of Japan, March, 2000. [written in Japanese]
- AEC (2000b): On the Treatment and Disposal of Transuranic Radionuclide Bearing-Radioactive Waste, Atomic Energy Commission of Japan. [written in Japanese]
- AEC (2000c): Long-term Program for Development and Utilization of Nuclear Energy, Atomic Energy Commission of Japan. [written in Japanese]
- ANDRA (2001): Dossier 2001 Argile (Progress Report on Feasibility Studies & Research into Deep Geological Disposal of High-level, Long-lived Waste).
- ANRE (2004): Analyses and Evaluations of Cost Structure of the Backend Project and Benefits of Nuclear Power Plants, Agency for Natural Resources and Energy. [written in Japanese]
- BMU (2001): Government decree on protection from ionized radiation, Bundesministerium für Forschung und Technologie.
- Chikasui Handobukku Henshu-iinkai (1979): Chikasui Handobukku (in Japanese) (Groundwater Handbook, translated by JAEA), Kensetsu Sogo Chosakai. [written in Japanese]
- DOE (2002): Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada.
- DSIN (1991): Basic law on Safety No. III.2.f, "Geological disposal of radioactive waste." [written in Japanese]
- HSK & KSA (1993): Protection Objectives for the Disposal of Radioactive Waste. HKS-R-21/e.
- IAEA (1995): The principle of radioactive waste management, IAEA Safety Fundamentals, Safety Series No. 111-F.
- IAEA (2004): Geological Disposal of Radioactive Waste, DRAFT SAFETY REQUIREMENTS, IAEA SAFETY STANDARDS SERIES DS154, Vienna.
- JNC (2000): H12: Project to Establish the Scientific and Technical Basis for HLW Disposal in Japan, Project Overview Report, Supporting Report 1, 2 and 3, Japan Nuclear Cycle Development Institute, JNC TN1410 2000-001, -002, -003 and -004.
- Nagra (1985): Projekt Gewähr 1985, Nagra Gewähr Report Series NGB 85-01/09.
- Nagra (1994): Kristallin-1 Safety assessment report, Nagra Technical Report NTB 94-10.
- Nagra (2002): Project Opalinus Clay; Safety Report; Demonstration of disposal feasibility for spent

- fuel, vitrified high-level waste and long-lived intermediate-level waste (Entsorgungsnachweis), Nagra Technical Report NTB 02-05.
- NRC (1982): 10 CFR Part61: Licensing Requirement for Land Disposal of Radioactive Waste.
- NRC (2002): 10 CFR Part63: Regulations for the Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada.
- 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), The Committee on Safety Regulations of Radioactive Waste, Nuclear Safety Commission of Japan, December, 1986. [written in Japanese]
- NSC (1987): Reference Radionuclide Concentration Values for Low-level Radioactive Solid Waste to be Land Disposed, Nuclear Safety Commission of Japan, June, 1987. [written in Japanese]
- NSC (1988): Fundamental Guidelines for Licensing Review of Land Disposal Facility of Low-level Radioactive Waste, Nuclear Safety Commission of Japan. [written in Japanese]
- NSC (1992): Reference Radionuclide Concentration Values for Low-level Radioactive Solid Waste to be Land Disposed (2nd Interim Report), Special Committee on Safety Standards for Radioactive Waste, Nuclear Safety Commission of Japan, February, 1992. [written in Japanese]
- NSC (2000a): Reference Radionuclide Concentration Values for Low-level Radioactive Solid Waste to be Land Disposed (3rd Interim Report), Special Committee on Safety Standards for Radioactive Waste, Nuclear Safety Commission of Japan, June, 2000. [written in Japanese]
- NSC (2000b): On the Safety Regulations for the Disposal of High-Level Radioactive Waste (1st Report), The Committee on Safety Regulations of Radioactive Waste, Nuclear Safety Commission of Japan, October, 2000. [written in Japanese]
- NSC (2003): On the Safety Regulations for the Shallow Disposal System of Radioactive Waste Generated from Facilities for Radioisotope Use (Proposal), Subcommittee on Radioactive Waste, Special Committee on the General Safety of Nuclear Energy, Nuclear Safety Commission of Japan, December, 2003. [written in Japanese]
- NSC (2004): Common Important Issues for Safety Regulations of Radioactive Waste, Special Committee on Radioactive Waste and Decommissioning Measures, Nuclear Safety Commission of Japan, June, 2004. [written in Japanese]
- OECD/NEA (2004): POST-CLOSURE SAFETY CASE FOR GEOLOGICAL REPOSITORIES. OECD/NEA Nuclear Energy Agency, Paris, France, 2004.
- PNC (1992): Research and Development on the Geological Disposal of High-Level Radioactive Waste (H3), Power Reactor and Nuclear Fuel Development Corporation, PNC TN1410

92-081. [written in Japanese]

RSA93 (1997): Disposal Facilities on Land for Low and Intermediate Level Radioactive Wastes: Guidance on Requirements for Authorisation, Environment Agency, Scottish Environmental Agency and Department of the Environment for Northern Ireland.

SKB (1999): SR-97 – Post-closure safety. Main report, Vol. II, SKB Technical Report TR-99-06.

SSI (1998): The Swedish Radiation Protection Institute's Regulations on the Protection of Human Health and the Environment in connection with the Final Management of Spent Nuclear Fuel and Nuclear Waste. SSI FS 1998:1.

SSI (2000): The Swedish Radiation Protection Institute's Regulations Concerning Final Management of Spent Nuclear Fuel and Nuclear Waste with Background and Comments. SSI Report 2000:18.

STUK (2001): Long-term Safety of Disposal of Spent Nuclear Fuel. YVL, 8.4.

STUK (2003): Disposal of Reactor Waste. YVL, 8.1.

TRU Coordination Team (2000): Outline of TRU waste disposal, Federation of Electric Power Companies, JNC TY1400 2000-001, TRU TR-2000-01. [written in Japanese]