DEVELOPMENT OF PROBABILISTIC ASSESSMENT METHODOLOGY FOR GEOLOGIC DISPOSAL OF RADIOACTIVE WASTES

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Abstract

The probabilistic assessment methodology is essential to evaluate uncertainties of long-term radiological consequences associated with geologic disposal of radioactive wastes. We have developed a probabilistic assessment methodology to estimate the influences of parameter uncertainties/variabilities. An exposure scenario considered here is based on a groundwater migration scenario. A computer code system GSRW-PSA thus developed is based on a non site-specific model, and consists of a set of sub-modules for sampling of model parameters, calculating the release of radionuclides from engineered barriers, calculating the transport of radionuclides through the geosphere, calculating radiation exposures of the public, and calculating the statistical values relating the uncertainties and sensitivities. The results of uncertainty analyses for α-nuclides quantitatively indicate that natural uranium (^{238}U) concentration is suitable for an alternative safety indicator of long-lived radioactive waste disposal, because the estimated range of individual dose equivalent due to ^{238}U decay chain is narrower than that due to other decay chain (^{237}Np decay chain). It is internationally necessary to have detailed discussion on the PDF of model parameters and the PSA methodology to evaluate the uncertainties due to conceptual models and scenarios.

1. INTRODUCTION

The Japan Atomic Energy Research Institute (JAERI) has developed the deterministic safety assessment code system GSRW (Generic Safety assessment code for geologic disposal of Radioactive Waste) [1], that is based on a normal evolution scenario. The GSRW, in which modular type of source term models, geosphere models and a biosphere model are interlinked, intends to evaluate radiological consequences to an individual or a population due to radionuclides released from geologic radioactive waste repositories in a deep stable rock mass. This kind of integrated code system is essential to construct probabilistic assessment methodologies which intend to evaluate uncertainties associated with assessment, as were developed by AECL (SYVAC) [2] and JRC-ISPRA (LISA) [3, 4, 5], while both code systems employ simpler sub-models than those used in the GSRW. We have developed a new probabilistic safety assessment code system GSRW-PSA by conjunction of the GSRW code, parameter sampling code and statistical analysis routine which evaluates uncertainties and sensitivities. The GSRW-PSA can handle parameter uncertainties in the source term models and geosphere models by using the Monte Carlo PSA (Probabilistic Safety Assessment) technique.This paper summarizes a probabilistic safety assessment methodology based on the Monte Carlo simulation technique, and the results of uncertainty analyses for geologic disposal of high-level radioactive waste (HLW) using the computer code system GSRW-PSA.

2. METHODOLOGY

2.1. Assessment Scenario

The assessment scenario considered here is based on a groundwater migration scenario, assuming that the performance of the disposal system is not affected by probabilistic events as is the case of normal

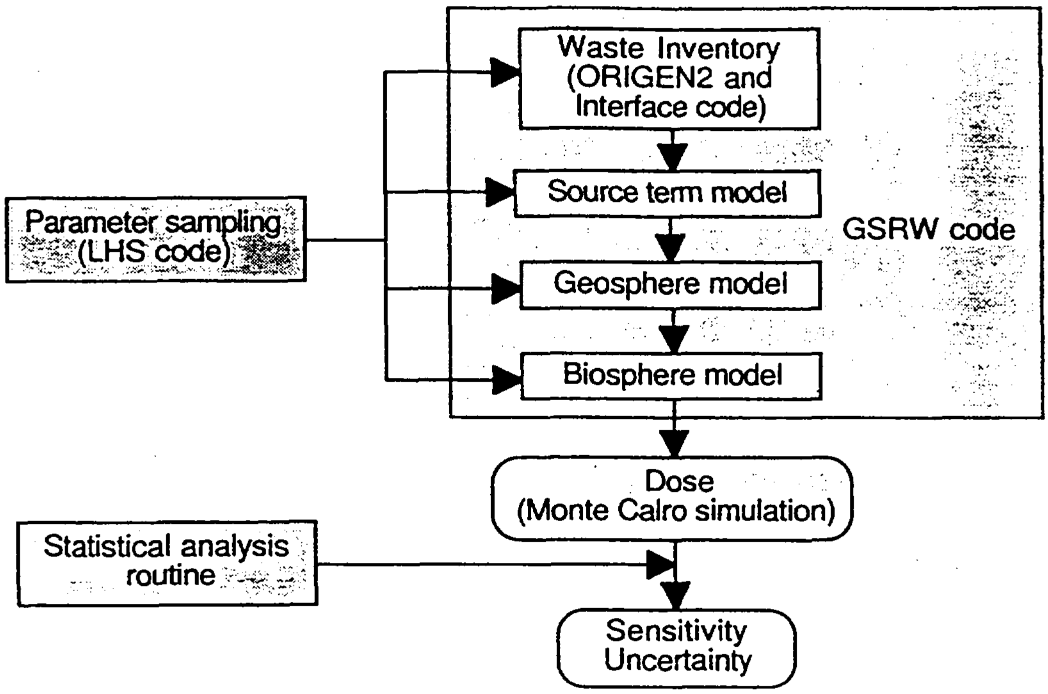

Fig. 1 Flow chart of GSRW-PSA code system

evolution scenario. It is assumed in the scenarios that all of the components involved in the repository are resaturated eventually with groundwater, after the closure of the repository. Degradation, with groundwater thus contacted, of the components occurs, which results in corrosion of the container and then dissolution of the vitrified matrix. These processes lead to the release of radionuclides into buffer material. Since the hydraulic conductivity in the zone may negligibly low, the transport in the buffer material is mainly controlled by the diffusion mechanism. The subsequent transport in the geosphere is governed by groundwater flow through fractured porous media. The control processes of the transport are the advection, dispersion including molecular diffusion and mechanical dispersion, the retention with mineral components of a rock, and the radioactive decay. Radionuclides entered into adjacent aquifers are diluted with a large volume of groundwater and further by surface water bodies.

2.2. Submodels and PSA method

The GSRW is composed of four interlinked models, ORIGEN2 [6] and interface codes, source term models, geosphere models and a biosphere model, as illustrated in **Fig. 1.** The ORIGEN2 and interface codes are used to evaluate the inventory of radionuclides in HLW as a function of time. The second models evaluate fluxes of radionuclides from a disposal facility which consists mainly of a vitrified waste form, a metallic container and buffer material. Two kinds of source term models are provided: Model-1 which simulates the dissolution of silicate component of glass and assumes that the radionuclides in the vitrified wastes are released in proportion to the leaching rate of silicate component, and Model-2 which assumes that the concentration of a radionuclide is limited by the solubility of its specific chemical form at the interface between the buffer and the waste. The third model analyzes the transport of radionuclides in the geosphere, which is based on analytical or numerical solutions of a mass transport equation involving an one-dimensional advection, a three or one-dimensional dispersion, a linear sorption and a decay chain. The three-dimensional dispersion and one-dimensional advection model is used for analyses of mass transport in the rock mass, and the one-dimensional dispersion and advection model is used for analyses in the fractured-zone. The fourth model assesses the transport of radionuclides in the biosphere and the resulting radiological consequences to the man, which is based on a dynamic compartment model for the biosphere and a dose factor method for dose calculation. In the GSRW-PSA, a simplified biosphere model are used for a drinking of well water, to attach importance to the uncertainties arise from engineered and natural barriers.

The GSRW-PSA consists of a parameter sampling code, the GSRW code and a statistical analysis routine, as shown in **Fig. 1**. The parameter sampling code used here is Latin Hypercube Sampling (LHS) code [7] which was developed by the Sandia National Laboratories. This code generates a set of parameter values for the Monte Carlo simulation based on the specified Probability Density Functions (PDF) of parameters. Based on the results of Monte Carlo simulation using GSRW, the statistical analysis code evaluates the following statistical results as the uncertainty analysis:
- Scatter plot of peak dose values,
- Histogram of peak dose values,
- Cumulative Distribution Function (CDF) and Complementary Cumulative Distribution Function (CCDF) of peak dose values,
- Time dependent values of arithmetic mean, 5, 50 and 95 percentile values.

As the sensitivity analysis, this code evaluates the following statistical values:
- Peason's correlation coefficient,
- Spearman's rank correlation coefficient,
- Standardized Regression Coefficient (SRC),
- Standardized Rank Regression Coefficient (SRRC),
- Partial Correlation Coefficient (PCC),
- Partial Rank Correlation Coefficient (PRCC),

for each of the sampled parameters against peak dose values. This code also displays various figures of the above statistical results.

3. UNCERTAINTY ANALYSIS

3.1 Basic Assumption and Input Data

The GSRW intends to evaluate the potential radiological consequences of geologic disposal of HLW is based on a generic approach rather than site specific one, reflecting from the current situation in Japan. However, it might be required to define the disposal system to some extent so as to enable the analysis of transport of radionuclides in geologic formations and the biosphere following entry of groundwater into a repository and release of them to surrounding strata. A crystalline bedrock is now considered to be one of potential geologic strata for geologic disposal of HLW in Japan. We thus assume that a repository will be constructed in a deep and stable granite bedrock at one of depths ranging from 500 to 1,000 m.

The rock is described in terms of two major hydraulic units; fractured-zones and rock mass. It might be reasonable to assume that the repository is constructed in a stable rock mass surrounded by local fracture-zones which further connect with regional fracture-zones, as shown in **Fig. 2**, in order to avoid the occurrence of a short circuit of groundwater from the repository to the biosphere. As the stable

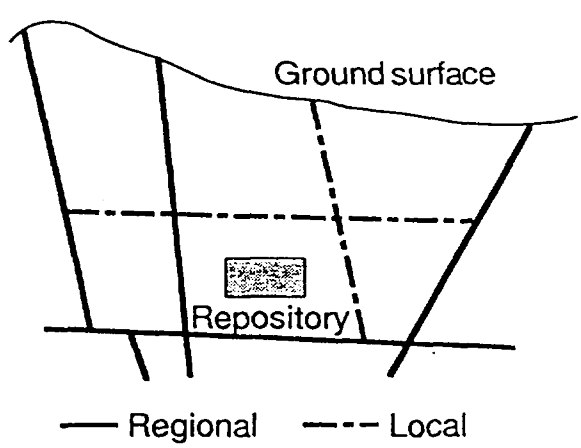

Fig. 2 Location of a potential HLW repository and major fracture-zones

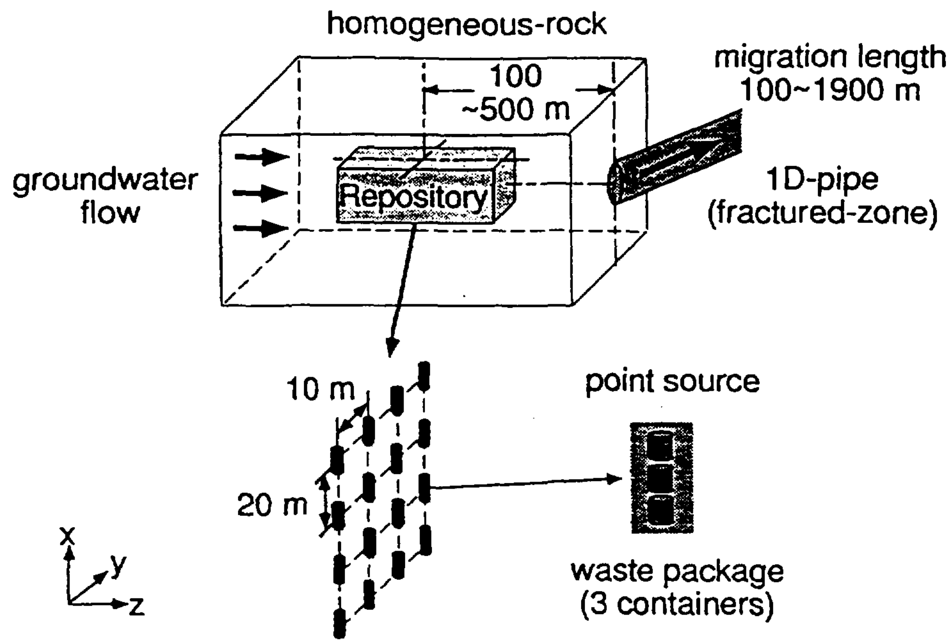

Fig. 3 Conceptualized HLW disposal system for the uncertainty and sensitivity analysis

rock mass has no major fracture, we can assume that the rock mass is homogeneous, while the scale of the stable rock mass is depend on the specific site. The radionuclide transport in the rock mass is simulated by the one-dimensional advection and three-dimensional dispersion model. The transverse dispersion was introduced to examine the effect of spatial array of waste packages. The radionuclide transport through the fractured-zone modeled as a pipe is analyzed by the one-dimensional advection and dispersion model.

Figure 3 shows the conceptualized disposal system assumed only for the uncertainty and sensitivity analysis. The analyses were made on radiologically important α-nuclides by using Model-2. The migration length of a homogeneous-rock surrounding a disposal facility was assumed to be in the range from 100 to 500 m, and that of a fractured-zone connecting to surface water bodies was in the range from 100 to 1,900 m. The facility was simulated by a simple vertical array of 16 point sources as shown in Fig. 3, each of which involves 3 containers (corresponding to 4 MTU of the spent fuels), to examine the effect of distance between waste packages on the individual dose equivalent due to the exposure pathway ingesting drinking water. In this analysis, 18 model parameters were selected to evaluate the parameter uncertainties of source term and geosphere models. Major input data used for the uncertainty and sensitivity analysis are shown in **Table 1**, and **Table 2** shows element specific data. Hydrology data used are based on the typical Japanese hydrogeologic condition, and geochemistry data were cited from SKI [8] and SKB [9] reports.

3.2 Results and Discussions

Figure 4 and 5 show the CCDFs of peak doses due to ^{237}Np and ^{238}U decay chains, respectively, and the Monte Carlo simulations are based on 5,000 runs. The 5 percentiles of peak doses due to ^{237}Np decay chain are 7.7E-11 (^{237}Np), 2.2E-10 (^{233}U), 6.9E-10 (^{229}Th) Sv/y, and the 95 percentiles are 4.1E-24 (^{237}Np), 1.7E-22 (^{233}U), 6.4E-22 (^{229}Th) Sv/y, respectively. The 90% confidence intervals (5 percentile - 95 percentile) of peak doses due to ^{237}Np decay chain have ranges of more than 12 figures. It means that the uncertainties of peak doses due to ^{237}Np decay chain are large, because the probable travel time of ^{237}Np to the biosphere in the disposal system is larger than its half-life. It is also clear from the results of sensitivity analysis that the groundwater flow velocity and the distribution coefficients (Kd) in the homogeneous-rock are very sensitive to the peak doses, which are not described here.

On the other hand, the 5 percentiles of peak doses due to ^{238}U decay chain are 3.0E-11 (^{238}U), 3.4E-11 (^{234}U), 8.4E-11 (^{230}Th), 1.7E-9 (^{226}Ra) Sv/y, and the 95 percentiles are 5.8E-14 (^{238}U), 6.6E-14 (^{234}U),

Table 1 Major input data for the uncertainty and sensitivity analysis

sub-model	parameter	distribution type	Min	Max
source term	life-time of container (y)	lognormal	1E3	1E5
	thickness of buffer (m)	loguniform	0.05	5.0
	porosity of buffer (-)	normal	0.2	0.4
	solubility limit (mol/L)	loguniform	see **Table 2**	
	diffusion coefficient of buffer (m^2/s)	lognormal	1E-12	1E-9
	Kd of buffer (m^3/kg)	lognormal	see **Table 2**	
homogeneous -rock	migration length (m) *	uniform	100	500
	velocity (m/y)	lognormal	0.01	1
	dispersion length (m) * (transverse)	uniform	1	5
	dispersion length (m) * (longitudinal)	uniform	10	50
	Kd (m^3/kg)	lognormal	see **Table 2**	
fractured -zone	migration length (m) *	uniform	100	1900
	velocity (m/y)	lognormal	0.1	10
	dispersion length (m) *	uniform	10	190
	Kd (m^3/kg)	lognormal	see **Table 2**	

* migration length and dispesion length are correlated each other.

Table 2 Element specific input data for the uncertainty and sensitivity analysis

Parameter		Element			
		Ra	Th	U	Np
solubility limits (mol/L)	Max	-	2E-8	1E-5	1E-8
	Min	-	2E-10	4E-8	1E-10
Kd (m^3/kg) buffer material	Max	-	1.0	1.0	1.0
	Min	-	0.002	0.01	0.01
Kd (m^3/kg) homogeneous-rock	Max	1.0	5	5	5
	Min	0.001	0.001	0.001	0.01
Kd (m^3/kg) fractured-zone	Max	0.1	5	0.1	0.1
	Min	0.001	0.001	0.001	0.001

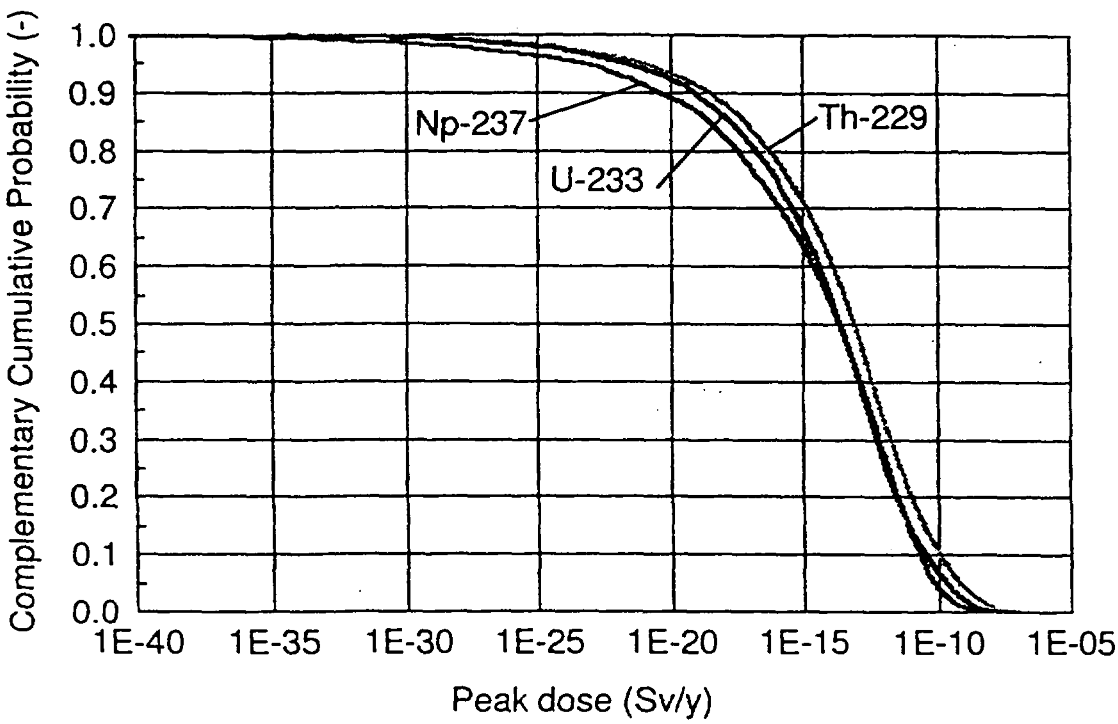

Figure 4 The CCDFs of peak doses due to ^{237}Np decay chain (Number of runs: 5,000)

Dose of Np-237 (Sv/y) ; 5 percentile: 7.7E-11, 95 percentile: 4.1E-24
Dose of U-233(Sv/y) ; 5 percentile: 2.2E-10, 95 percentile: 1.7E-22
Dose of Th-229 (Sv/y) ; 5 percentile: 6.9E-10, 95 percentile: 6.4E-22

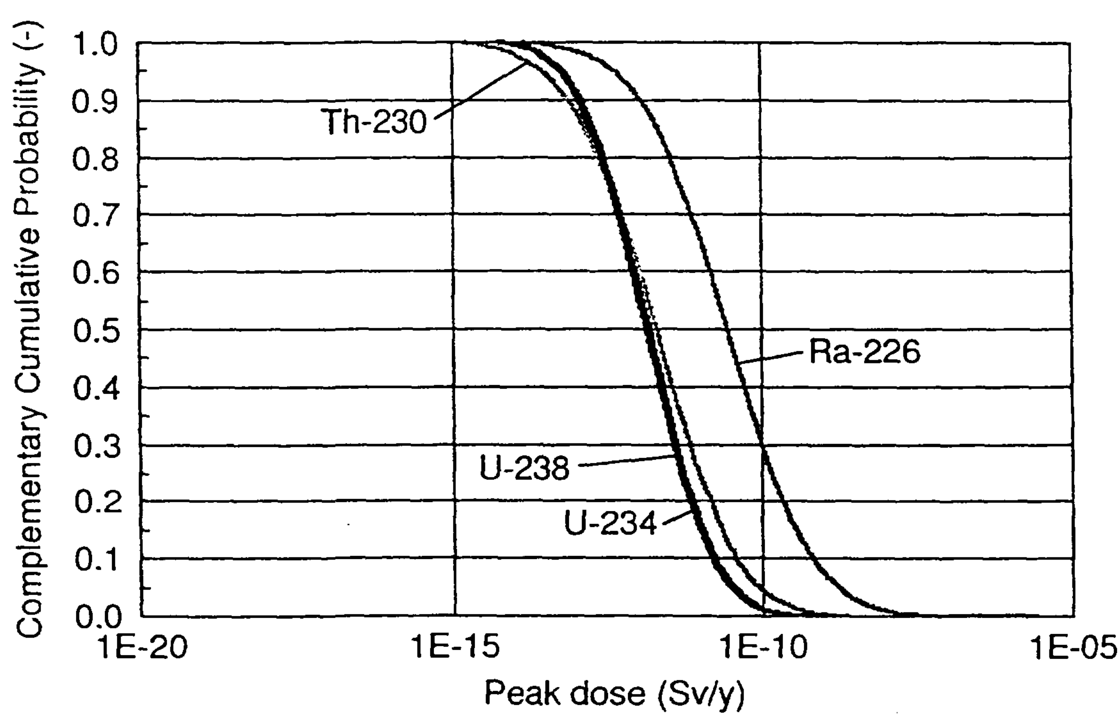

Figure 5 The CCDFs of peak doses due to ^{238}U decay chain (Number of runs: 5,000)

Dose of U-238 (Sv/y) ; 5 percentile: 3.0E-11, 95 percentile: 5.8E-14
Dose of U-234(Sv/y) ; 5 percentile: 3.4E-11, 95 percentile: 6.6E-14
Dose of Th-230 (Sv/y) ; 5 percentile: 8.4E-11, 95 percentile: 3.0E-14
Dose of Ra-226 (Sv/y) ; 5 percentile: 1.7E-09, 95 percentile: 4.3E-13

3.0E-14 (^{230}Th), 4.3E-13(^{226}Ra) Sv/y, respectively. The 90% confidence intervals of peak doses due to ^{238}U decay chain have ranges of about only 3~4 figures. The results indicate that the uncertainties of peak doses due to ^{238}U decay chain are relatively quite small because of its very long half-life, compared with those due to ^{237}Np decay chain. These results will not be changed significantly, except for a case the groundwater flow velocity in the homogeneous-rock is extreme low, which is too optimistic. Therefore, we can adequately choose natural uranium concentration for an alternative safety indicator of long-lived radioactive waste disposal besides the dose equivalent and risk.

4. CONCLUSIONS

We have developed the probabilistic safety assessment methodology (GSRW-PSA) to estimate the influences of parameter uncertainties. The results of uncertainty analyses for α-nuclides indicate that natural uranium (^{238}U) concentration is suitable for an alternative safety indicator of long-lived radioactive waste disposal quantitatively. It is internationally necessary to have detailed discussion on the PDF of model parameters for the parameter uncertainty analysis. The PSA methodology to evaluate the uncertainties due to conceptual models and scenarios is still under developing in any country. This issue also should be discussed internationally.

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