# Some Illustrative Examples from Parametric Studies for a Reference HLW Repository Using ACGEO

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## 1. Introduction

Spent fuels are to be directly disposed of and the only type of HLW in Korea. According to the basic repository design[1], which is similar to Swedish KBS-3 repository concept[2], spent fuel is encapsulated in corrosion resistant canisters, and then emplaced into the deposition holes surrounded by high density bentonite clay in tunnels at a depth of ~500 m in a plutonic rock (Fig. 1). The objective of this paper is to show how we could show reliability of radiological safety assessment through probabilistic calculation with parameters having uncertainties associated with safety of HLW system. To demonstrate a typical way, possibly among many others, some probabilistic calculation results for nuclide release and transport in the near- and far-field of the repository when such selected key parameters as buffer thickness, nuclide travel length in the fractured rock medium, and initial number of defective canisters are varying are presented. ACGEO, a case file developed by utilizing AMBER[3], a compartment modeling software package was used for calculation.

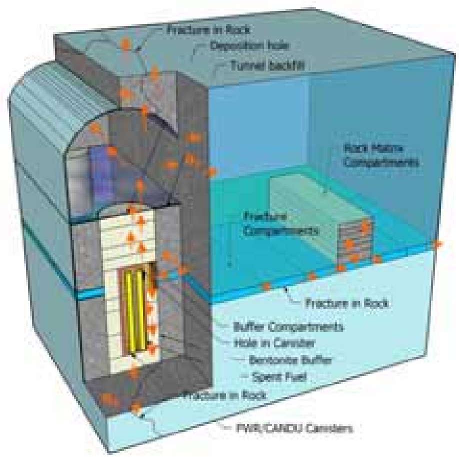



Fig. 1. Schematic near- and far-field system domain.

## 2. Modeling

Such near-field barriers as canister, surrounding buffer, and excavation damaged zone as well as far-field components including host rock and outer tunnel part are modeled as independent compartments as shown in Fig. 1. accounting for the geometry and materials through which nuclides released from the canister are transferred and transported. Quantitative estimation of nuclide release is made considering the material balance over each compartment to the other connected compartments (Fig. 2.). As the final step of safety assessment, evaluation of doses to human beings in the biosphere due to nuclides released from near- and far-field media and through the various pathways is made using the flux to dose conversion factors for converting nuclide flux to dose exposure rate, which is also calculated through a separate mathematical compartment model considering their decay chains among the compartments in biosphere adopting farming exposure pathway[4-5].

The material balance in any compartment for mass of nuclide, $N_i$ is represented as

$$\frac{dN_i}{dt} = \sum_{j \neq i} \lambda_{ji} N_j + \lambda^M M_i + S_i(t) - \sum_{j \neq i} \lambda_{ij} N_i - \lambda^N N_i \qquad (1)$$

where $\lambda$ = decay constant, $M$ = mass of parent nuclide, and $S$ = source/sink term in each compartment $i$. Nuclide flow rate from one compartment to the adjacent compartment is proportional to a mass transfer coefficient, $\lambda_{ij}$. Nuclides released from canisters with small holes (which is considered as initial intrinsic defect) are modeled to be ready to transport through surrounding near- and far-field media to finally reach to biosphere where nuclides give rise to doses to human beings. Once nuclides in the spent fuel matrix as well as in such gap portion as grain boundaries and cladding, where nuclides are immediately available to release are contacted with groundwater, their transfer and transport begin to take place.

Released through the initial canister hole from the spent fuel matrix and gap portion, nuclides then continue to transport surrounding buffer and tunnel backfill where diffusive transport is assumed to dominantly take place due to their low permeability. However in case nuclides meet groundwater bearing fractures in the surrounding host rock advective transport could also occur. Matrix diffusion into the stagnant groundwater in the rock matrix pores as well as sorptions onto both the fracture wall and matrix surfaces are also accounted for.[6]

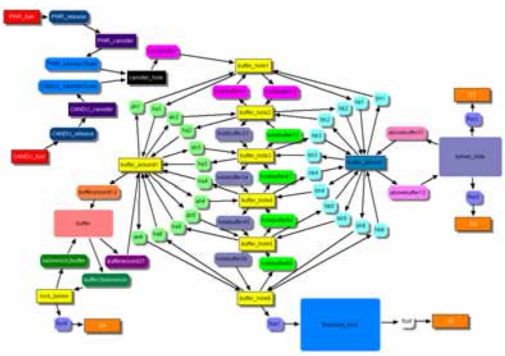
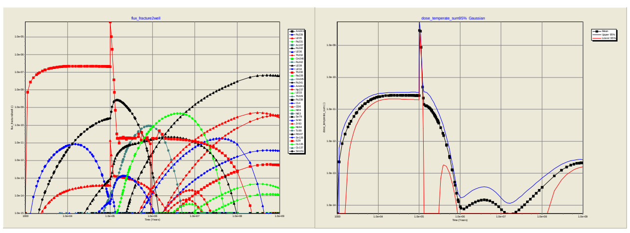


Fig. 2. Compartment modeling for nuclide flux calculation.

## 3. Illustration

Just for illustrative purpose, examples of quantitative result of far-field flux and individual dose due to released nuclides from varying travel distance in the fracture are shown (Figs.3 and 4), the first figure of which shows deterministic case for the nuclide flux to well, and the rest of which are for impact of probabilistically varying fracture length. All the values of input parameters and their distribution are assumed and/or taken reasonably from various sources to calculate and illustrate quantitative results.



(a) (b)

Fig. 3. (a) Mean nuclide fluxes to well and (b) mean total dose (fracture length~Triangular distribution(50, 100, 1000m) with 95% confidence limit.

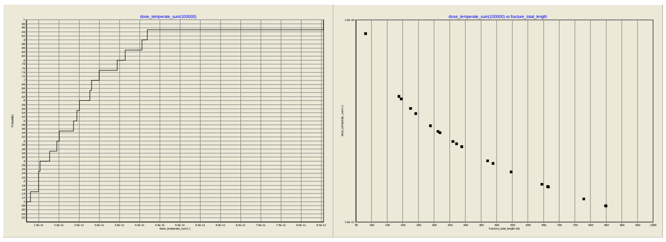


Fig. 4. (a) Calculated cdf for total dose at 100k yrs and (b) scatter plot for fracture length vs total dose at 100k yrs.

## 4. Concluding Remarks

Parametric studies for selected key parameters have been illustrated. Such calculation is also useful and necessary for further in-depth feedback to refine repository concept and then to build in an appropriate safety margin. ACGEO seems to be a useful tool to model such complicated system as HLW repository and to calculate the transient nuclide chain transport in the near- and far-field of the repository both on deterministic and probabilistic bases.

## REFERENCES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