NUCLIDE TRANSPORT CALCULATION IN NEAR- AND FAR-FIELD OF A REFERENCE HLW REPOSITORY USING AMBER

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ABSTRACT

In Korea currently 16 PWRs and 2 CANDUs are in operation, from which the cumulative amount of spent nuclear fuels of approximately 5,788MTU has been generated by the end of 2002 and is being stored at each NPP site. Spent fuels will be directly disposed of and the only type of HLW in Korea. The relevant R&D program for disposal of high-level radioactive waste has been carried out at Korea Atomic Energy Research Institute (KAERI) since early 1997 in order to develop a concept of Korea Reference Repository System (KRR) by the end of 2007 for direct disposal of nuclear spent fuel. A preliminary reference geologic repository concept considering such established criteria and requirements as waste and generic site characteristics in Korea was roughly envisaged in 2003 focusing largely on the near-field components of the repository system. According to the basic repository concept, which is similar to that of Swedish KBS-3 repository, the spent fuel is encapsulated in corrosion resistant canisters, even though the material has not yet been determined, and then emplaced into the deposition holes surrounded by high density bentonite clay in tunnels constructed at a depth of about 500 m in a stable plutonic rock body. The objective of this paper is not only to demonstrate how much a reference repository is safe in the generic point of view with several possible scenarios and cases associated with a preliminary repository concept by conducting calculations for nuclide release and transport in the near- and far-field components of the repository using AMBER, a compartment modeling software package, even though that much enough site specific information has not been available yet but also to show a possibility by which a generic safety assessment could be performed for further developing Korea reference repository concept. To this end some modeling scheme and numerical illustration are presented.

INTRODUCTION

The HLW-relevant R&D program for disposal of high-level radioactive waste has been carried out at Korea Atomic Energy Research Institute (KAERI) since early 1997, from which a conceptual KRR for direct disposal of nuclear spent fuel is to be introduced by the end of 2007. A preliminary reference geologic repository concept considering such established criteria and requirements as spent fuel and generic site characteristics in Korea was roughly sketched in 2003 [1]. According to basic repository concept, which is much similar to Swedish KBS-3 repository, nuclide release from the near- and far-field system has been investigated for the initial canister defect scenario which was also similarly done through SR97 study [2].

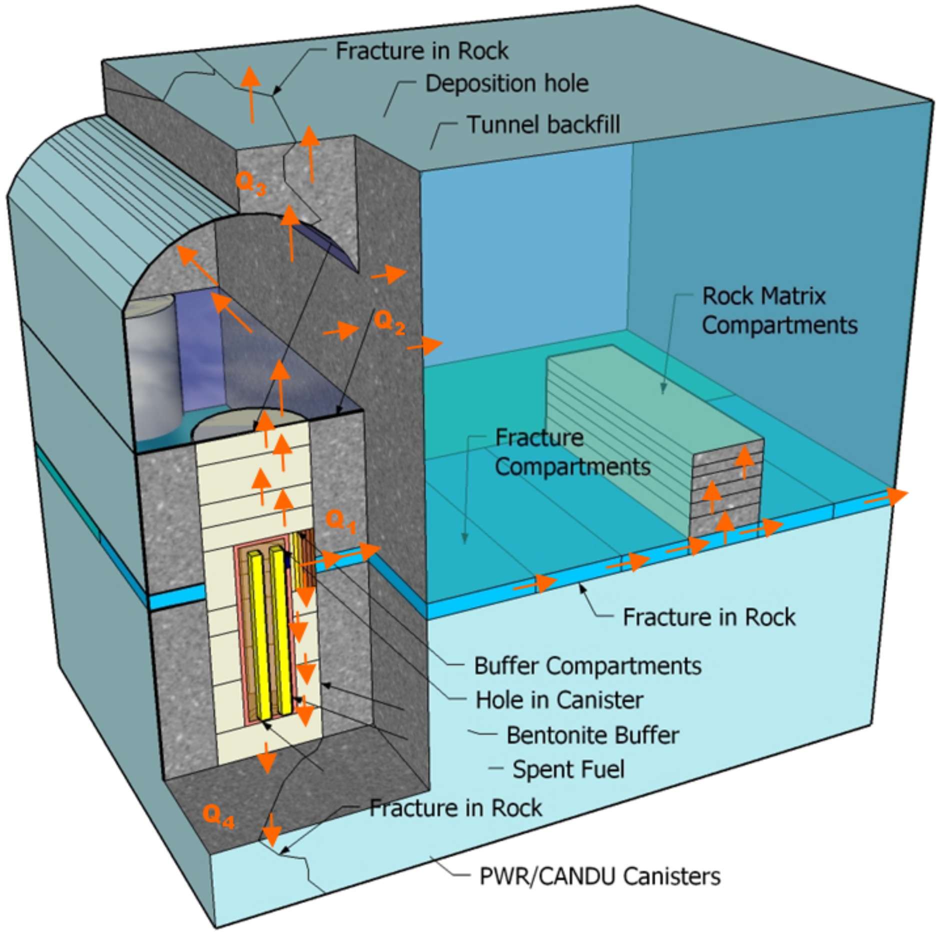


Fig. 1. Schematic near- and far-field system domain.

To demonstrate how much a reference repository is safe in the generic point of view with several possible scenarios and cases associated with a preliminary repository concept by conducting calculations for nuclide release and transport in the near- and far-field and also even biosphere components of the repository, although sufficient information has not been available that much yet, nuclide release calculation for various pathways is mandatory to show an appropriate methodology by which both a generic and site-specific safety assessment could be performed for further development of Korea reference repository concept.

For normal case all the values of input parameters were assumed and taken reasonably from various sources to calculate quantitative results and to illustrate the difference from the results for some other varying cases.

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 calculated by utilizing AMBER compartment modeling package [3] considering the material balance over each compartment to the other connected compartments (Fig. 2.).

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 the final step of safety assessment. To calculate the flux to dose conversion factors for converting nuclide flux to dose exposure rate, a separate mathematical compartment model has been constructed considering material balance with their decay chains among the compartments in biosphere as shown in Fig. 3.

Modeling

Nuclide Flux Calculation

Such near-field barriers as canister, surrounding buffer, and excavation damaged zone as well as far-field geosphere components including fractured host rock and outer tunnel part of the repository are modeled as independent compartments accounting for their geometry and materials through which nuclides released from the canister are transferred and transported to the biosphere. The material balance in a 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: λ = decay constant, M = mass of parent nuclide, and S = source/sink term in each compartment *i*.
Magnitude of nuclide flow rate from one compartment to the adjacent compartment is proportional to a mass transfer coefficient, λ_{ij} which is defined, for example, by the diffusivity of the materials, the cross sectional areas used for transport and the diffusion lengths from the representative nodal point of the compartment to the interface.

Nuclides released from canisters with small holes, emplaced in the deposition holes surrounded by high density bentonite clay 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 as illustrated in Fig. 1.

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.

For general case assuming or taking reasonably all the values of input parameters from various sources several quantitative calculation and comparisons are made.

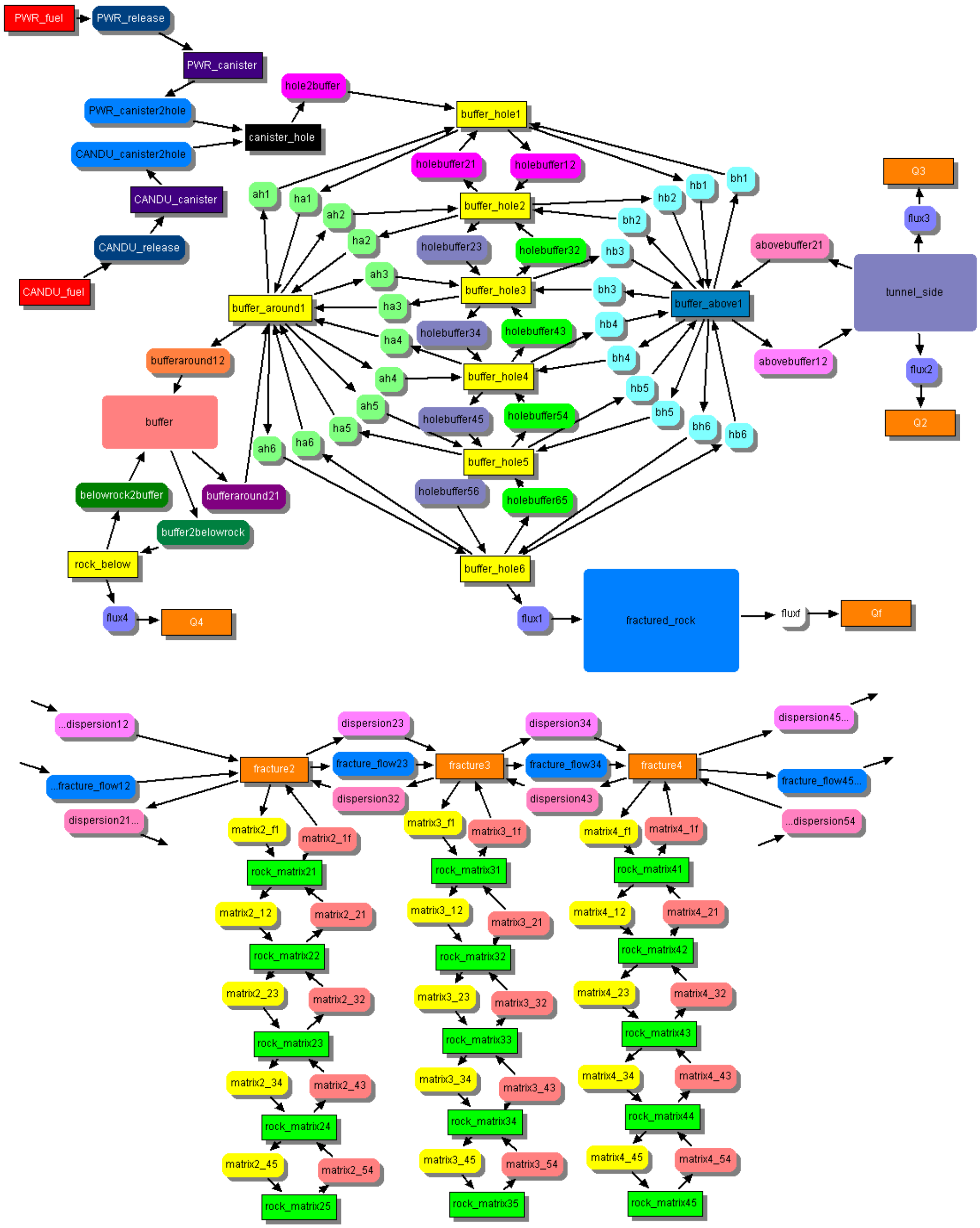


Fig. 2. Compartment modeling for nuclide flux calculation.

Most transfers between each near-field compartment can be specified in terms of electrical resistance analogy between adjacent two compartments which is straightforwardly given by resistance, $\Omega_{ij} = \frac{1}{2A_{ij}}\left(\frac{d_i}{D_e^i}+\frac{d_j}{D_e^j}\right)$ where A_{ij} = common area between the two compartment perpendicular to transport direction, D_e = effective diffusion coefficient, and d = length of the compartment in the direction of transport [4]. The associated transfer rate between compartments i and j is then given by $\lambda^{ij} = \frac{1}{\theta_i R_i V_i \Omega_{ij}}$ where θ_i =compartment porosity, R_i = retardation, and V_i = compartment volume.

Considering mass transfer rate between compartments for a sorbing nuclide as

$$\text{rate} = \left.\frac{\dot{Q}c}{N}\right|_{c=\frac{N}{\theta VR}} = \frac{\dot{Q}}{\theta VR} \tag{2}$$

where: $\dot{Q}$ = groundwater flux, N = nuclide mass or amount in compartment i and c = aqueous concentration and $N = \theta Vc + \rho Vs = \theta Vc\left(1+\frac{\rho K_d}{\theta}\right) = \theta VcR$ when assuming linear isotherm with distribution coefficient K_d where s = mass of nuclide adsorbed and precipitated on the solid per bulk dry mass of porous medium having density ρ .

Then diffusive flow rate from compartment i to compartment j can be expressed as

$$\text{rate}_{i\to j} \equiv \lambda_{ij} = \frac{\text{nuclide flux}}{\text{nuclide amount}} = \left.\frac{A_i D_e^i c_i / d_i}{N_i}\right|_{c=\frac{N_i}{\theta_i V_i R_i}} = \frac{A_i D_e^i}{\theta_i V_i R_i d_i} \tag{3}$$

Similarly introducing Peclet number, Pe, dispersion flow rate from i is expressed as

$$\text{rate}_{i\to j} \equiv \lambda_{ij} = \frac{\text{nuclide flux}}{\text{nuclide amount}} = \left.\frac{A_i D_i c_i / d_i}{N_i}\right|_{\substack{D_i=\frac{vL}{\text{Pe}} \\ c=\frac{N_i}{\theta_i V_i R_i}}} = \frac{A_i v_i L/\text{Pe}}{\theta_i V_i R_i d_i} \tag{4}$$

Especially for compartments in the fracture assuming $\theta_i = 1$,

$$\text{rate}_{i\to j} \equiv \lambda_{ij} = \frac{\text{nuclide flux}}{\text{nuclide amount}} = \left.\frac{A_i D_i c_i / d_i}{N_i}\right|_{\substack{D_i=\frac{vL}{\text{Pe}} \\ c=\frac{N_i}{\theta_i V_i R_i}}} = \left.\frac{A_i v_i L/\text{Pe}}{\theta_i V_i R_i d_i}\right|_{\substack{d_i=\Delta L \\ A_i/V_i \equiv \Delta L \\ \theta_i=1}} = \frac{v_i L/\text{Pe}}{R_i(\Delta L)^2} \tag{5}$$

Introducing resistances, $\Omega_{ij} = \frac{1}{2A_{ij}}\left(\frac{d_i}{D_e^i}+\frac{d_j}{D_e^j}\right)$, Eq. (5) can be written as

$$\text{rate}_{i\to j} = \lambda_{ij} = \left.\frac{A_{ij} D_e^{i\to j}}{\theta_i V_i R_i d}\right|_{\Omega_{ij}=\frac{1}{2A_{ij}}\left(\frac{d_i}{D_e^i}+\frac{d_j}{D_e^j}\right)} = \frac{1}{\theta_i R_i V_i \Omega_{ij}} \tag{6}$$

For compartment in contact with water flowing fractures in the rock, diffusive transport can be determined by introducing a fictitious equivalent flow rate, $\dot{Q}_{eqv}$, which is visualized as the groundwater flow rate that carries away nuclides with the concentration equivalent at the compartment interface, from which the flow resistance is obtained as $\Omega_{i,flowingwater} = \frac{1}{D\sqrt{2\pi A_{hole}}}$ [5] where A_{hole} = hole area. And also for the transfer into a fracture which is relatively much more narrow than buffer medium around the canister, by calculating the nuclide flow rate due to diffusion, transfer resistance at the mouth of the fracture can be obtained as $\Omega_{i,fracture} = \frac{\left(\frac{F_{x,0}}{b}\right)b}{DA_f}$ where $\left(\frac{F_{x,0}}{b}\right)b$ is as estimated by Neretnieks [6] where $\left(\frac{F_{x,0}}{b}\right) \approx 1 - 1.35\log {b}/{a} + 1.6\log\frac{d}{a}, 10^{-6} < {b}/{a} < 10^{-1}, 0.03 < {a}/{d} < 1$, A_f = fracture opening area contacting buffer, b = fracture half width, and a = canister length.

Dose Calculation

The final step of the safety assessment concerns nuclide transfer in the biosphere and the exposure pathways through which human being could be exposed. For modeling of huge uncertain system of biosphere as reasonably as possible, reference biosphere concept which has been developed as part of IAEA BIOMASS [7] has been adopted in this study. The nuclide transport processes and exposure pathways and associated input data derived from the separate study [8,9] has been incorporated into the compartment model shown in Fig. 3 which uses the same material balance of Eq. (1) and then implemented again using AMBER.

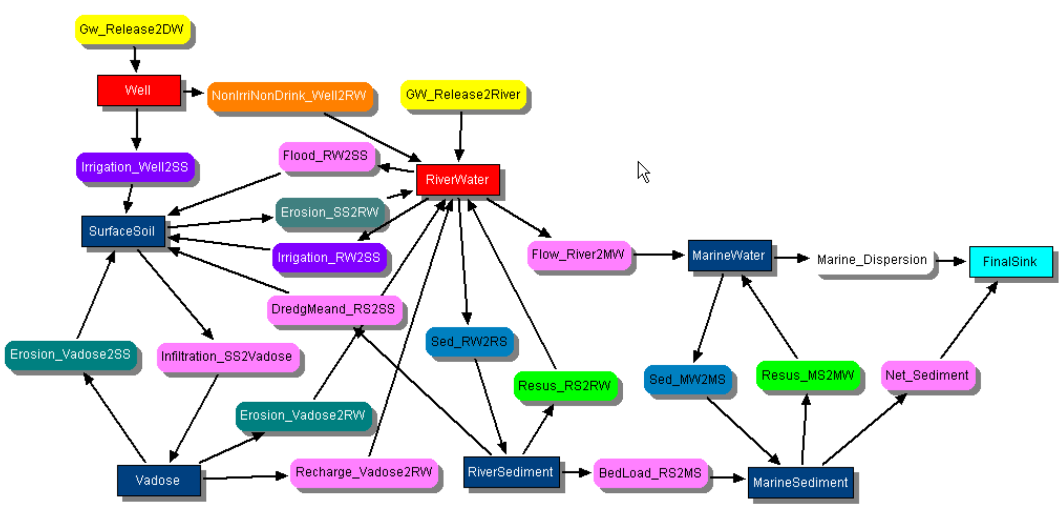


Fig. 3. Compartment modeling for dose conversion factor calculation.

Exposure pathways for groups of exposed individuals are classified into three cases as similarly done in H12 study [10]. The annual individual dose for each of internal exposure by ingestion, internal exposure by inhalation, and external exposure is calculated using the following equations based upon the nuclide concentration in each biosphere compartment:

Internal exposure by ingestion D_{ing} is expressed as

$$D_{ing} = DCF_{ing} ING\, C_{ing} \tag{7}$$

where: DCF_{ing} = the dose coefficient for ingestion [Sv/Bq], ING = the amount of ingested food [Kg/y] and C_{ing} = the nuclide concentration in the food [Bq/kg].
Similarly internal exposure by inhalation D_{inh} is also expressed as

$$D_{inh} = DCF_{inh} BR_{comp} O_{comp}\, C_{inh} \tag{8}$$

where: DCF_{inh} = the dose coefficient for inhalation [Sv/Bq], BR_{comp} = the inhalation rate [m3/h], O_{comp} = occupancy [h/y] and C_{inh} = the nuclide concentraton in dust [Bq/kg].
Finally an expression for external dose D_{ext} is

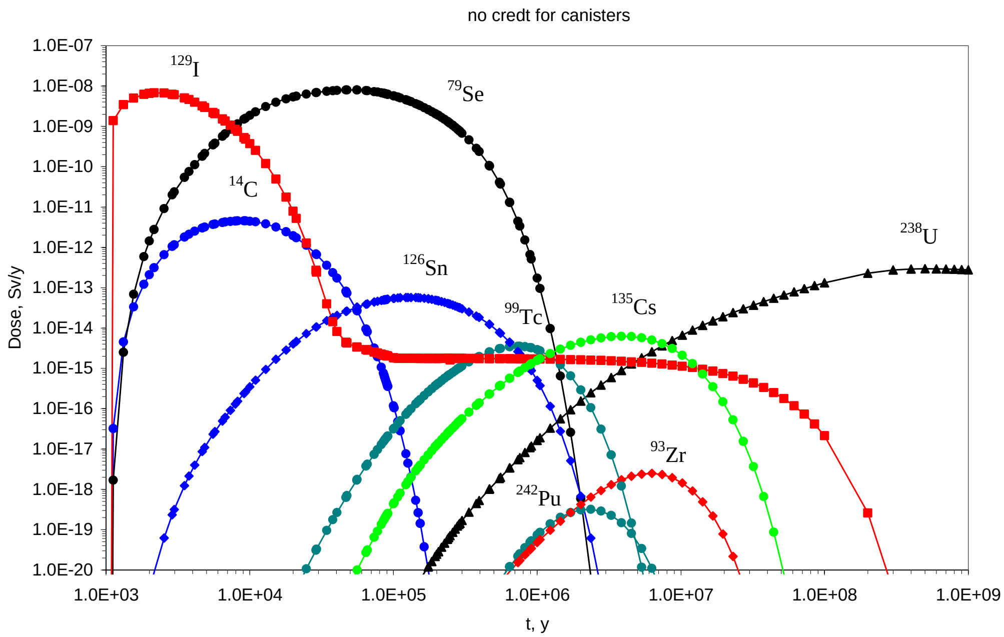
$$D_{ext} = DCF_{ext} O_{comp}\, C_{ext} \tag{9}$$

where: DCF_{ext} = the dose coefficient for external exposure [(Sv/h)/(Bq/m^3)] and C_{ext} = the nuclide concentration in water or soil [Bq/m^3].
Almost all of the data for the biosphere assessment only except for annual consumption rates have been adopted from the various literature including H12.

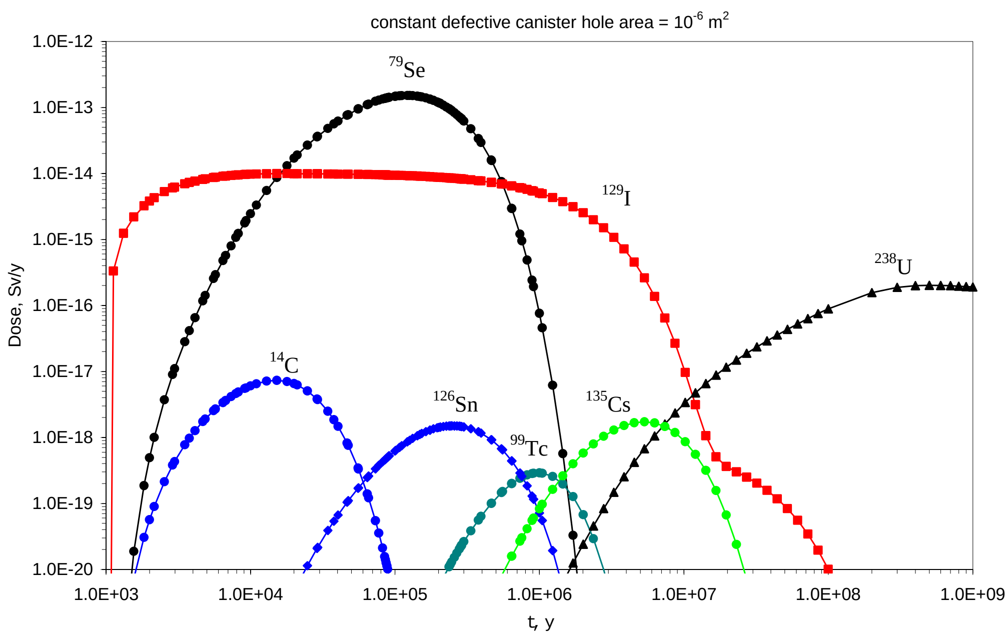
Illustration

Just for illustrative purpose, parts of quantitative estimation of individual dose due to released nuclides are shown in Fig.4, calculated with limited input data and for farming exposure pathway, the first one of which has no credit for canisters and the rest of which are for the cases that there are limited number of canisters having initial small inherent defective holes on them.

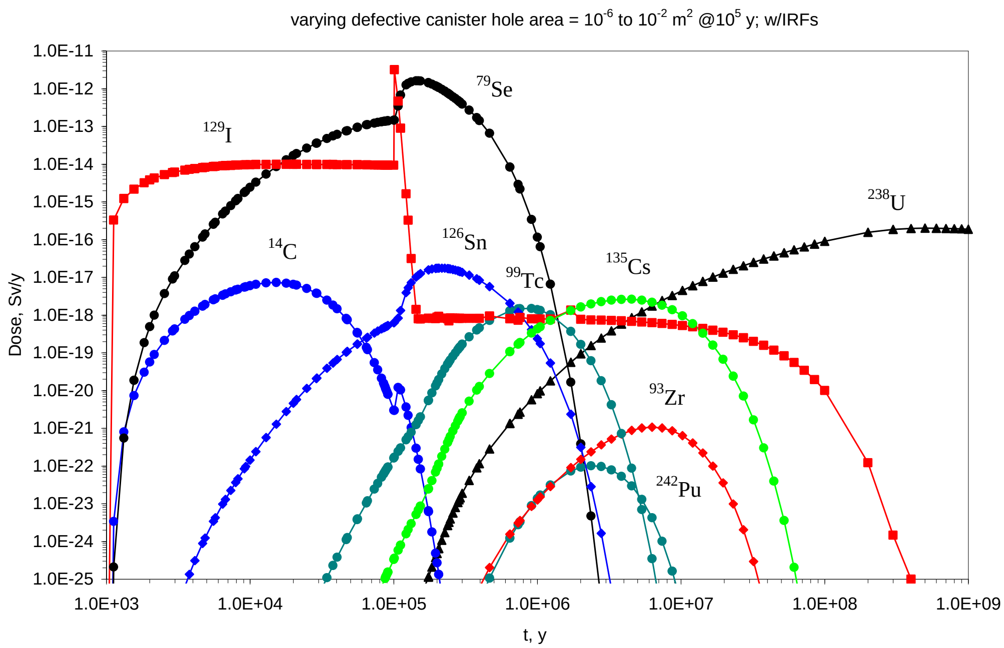
The breakthroughs altogether seem to show very typical behavior. In Fig. 4d, among others, showing the case that instant release from the gap portion of 10% of total initial inventory is not allowed for such nuclides as ^{14}C, ^{79}Se, ^{90}Sr, ^{99}Tc, ^{129}I, ^{135}Cs, and ^{137}Cs, the peaks of these nuclides are prominently lowered showing the large difference with the result in Fig. 4c. where the same instant release portion was allowed to all of them.

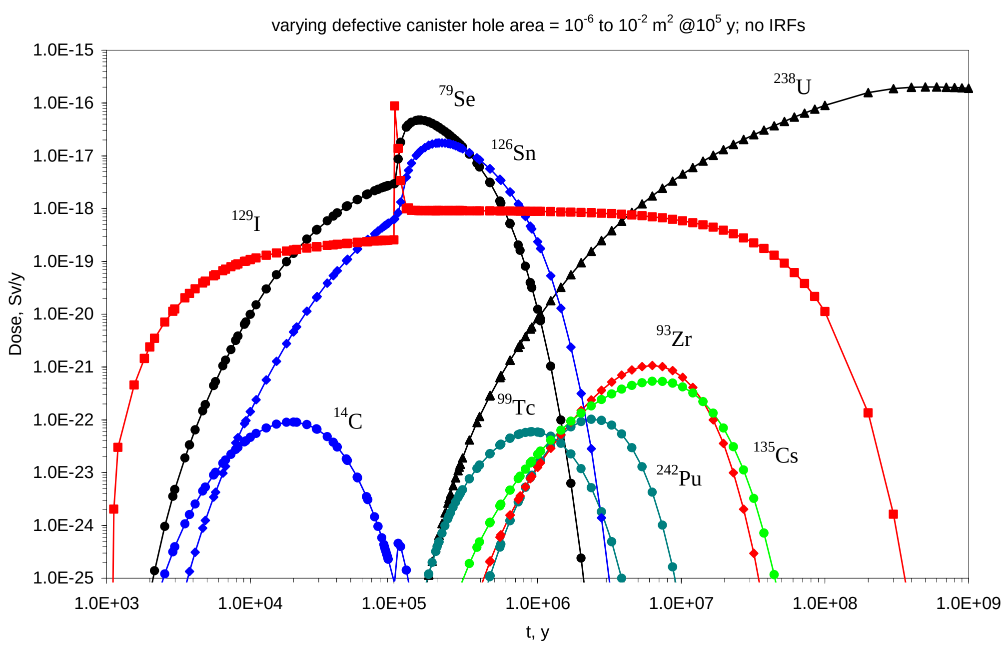

no credt for canisters
Dose, Sv/y
t, y
1.0E-07
1.0E-08
1.0E-09
1.0E-10
1.0E-11
1.0E-12
1.0E-13
1.0E-14
1.0E-15
1.0E-16
1.0E-17
1.0E-18
1.0E-19
1.0E-20
1.0E+03
1.0E+04
1.0E+05
1.0E+06
1.0E+07
1.0E+08
1.0E+09
129I
79Se
14C
126Sn
99Tc
135Cs
238U
93Zr
242Pu

(a)


constant defective canister hole area = 10-6 m2
Dose, Sv/y
t, y
1.0E-12
1.0E-13
1.0E-14
1.0E-15
1.0E-16
1.0E-17
1.0E-18
1.0E-19
1.0E-20
1.0E+03
1.0E+04
1.0E+05
1.0E+06
1.0E+07
1.0E+08
1.0E+09
79Se
129I
238U
14C
126Sn
135Cs
99Tc

(b)





(c)

(d)

Fig. 4. Calculated dose rates: (a) in case of no canister credit, (b) in case of constant initial canister hole area (c) in case of suddenly changing hole area at 10^5 years, and (d) in case instant release is not allowed.

CONCLUDING REMARKS

To demonstrate a way to assess the safety of a reference HLW repository in Korea in the generic point of view with possible scenarios and cases, calculations for nuclide release and transport in the near- and far-field and also biosphere components of the repository have been made and illustrated.

Even though sufficient information associated with reference repository has not been available yet, such release calculation for various pathways and scenarios is necessary to show an appropriate or possible methodology by which both a generic and site-specific safety assessment could be performed for further in-depth feedback to refine repository concept. AMBER which has not any own mathematical model in it 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 as well as biosphere system with many complicated exposure pathways although verification study should be followed and optimum criteria to avoid erroneous result and/or to improve results should be sought out by investigating optimized numerical conditions.

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