SIMPLIFIED SEMI-ANALYTICAL MODEL FOR MASS TRANSPORT SIMULATION IN UNSATURATED ZONE

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ABSTRACT

This paper describes a simple model to determine the flux of radionuclides released from a concrete vault repository and its implementation through the development of a computer program. The radionuclide leach rate from waste is calculated using a model based on simple first order kinetics and the transport through porous media bellow the waste is determined using a semi-analytical solution of the mass transport equation. Results obtained in the AIEA intercomparison program are also related in this communication.

Key Words: radioactive waste, safety assessment, repository, computer code.

I. INTRODUCTION

Safety assessment of near surface radioactive waste disposal repositories requires the development of computer codes in order to simulate the movement of the radionuclides through geosphere over time period as long as hundreds of thousands years. Several commercial and public domain codes can do that properly, in general using numerical solution of the flux and mass transport equations. In this process, however, limited site specific data are often available.

In this work, a simple model to determine the flux of the radionuclides released from a concrete vault repository is described. A computer program was developed to simulate the radionuclide released from waste form, transport through multiple engineered barrier layers of the repository and transport in the unsaturated zone. The methodology adopted requires a few input data and is not intended to be a predictive tool, but rather to provide a fast and robust tool to perform initial screening source term calculations. Also, the model performance in the IAEA NSARS intercomparison program is presented.

II. MODEL DESCRIPTION

The code was written in FORTRAN language and is currently being adapted for running in MS EXCEL spreadsheet.

Release Rate from Waste. Radionuclide release from the waste form is calculated using a simplified model based on simple first order kinetics. The leaching rate constant λ_l is determined by the following equation:

$$\lambda_l = \frac{V}{H\left(\theta + \rho\, K_d\right)} \qquad (1)$$

where V is the infiltration rate through the waste, H is the waste depth, θ is the moisture content and ρ is the waste bulk density.

Transport Through Vault Engineered Barrier Layers and Unsaturated Soil. Radionuclide transport was determined using analytical solution of the mass transport equation, considering the limiting case of unidirectional convective transport with three-dimensional dispersion in an isotropic medium. The transport equation is solved in terms of Green's functions, and applied successively in the multiple layers of the engineered barriers of the repository. Hence, values of flux calculated at the end of the each layer were used as input data for the next one.

Radionuclide concentration (C), at distance z from an horizontal area source of length L and width W, is therefore calculated by [1]:

$$C(t) = \frac{Q(\tau)}{\eta R_d} ZXY \ , \qquad (2)$$

where

$$Z = \frac{1}{\sqrt{4\pi D_z t/R_d}} \exp\left\{ -\frac{\left(z - ut/R_d\right)^2}{4D_z t/R_d} - \lambda t \right\} \qquad (3)$$

$$X = \frac{1}{2L}\left\{ erf\ \frac{\left(L/2 + x\right)}{\sqrt{4D_x t/R_d}} + erf\ \frac{\left(L/2 - x\right)}{\sqrt{4D_x t/R_d}} \right\} \qquad (4)$$

$$Y = \frac{1}{2W}\left\{ erf\ \frac{\left(W/2 + y\right)}{\sqrt{4D_y t/R_d}} + erf\ \frac{\left(W/2 - y\right)}{\sqrt{4D_y t/R_d}} \right\} \qquad (5)$$

and $Q(\tau)$ is the quantity released at time τ; η is the moisture content of the medium, R_d is the retardation coefficient; D_z, D_x and D_y are the dispersion coefficients; and u is the water pore velocity. For barrier layers, transverse dispersivity was not considered and $X = 1/L$, $Y = 1/W$. Water velocity u was assumed constant to allow analytical solution of the problem and was set as $u = u(\tau)$.

In considering continuous release, the total flux to the groundwater is evaluated by summing all contributions over the release period.

III. RESULTS OBTAINED IN THE NSARS INTERCOMPARISON STUDIES

The Test Case 2B of the NSARS programme was focused in the near-field modelling, considering a trench and a concrete vault facilities. The first facility is assumed to be a pit excavated in the native soil and the second one is an engineered concrete vault surrounded by layers of soil, sand and clay on the top and sand and soil at the bottom. The exercise was well specified in terms of engineered facility geochemical properties, radionuclide inventory, waste form, infiltration rate and site characterization. For results comparison purposes, calculation of maximum flux and concentration in the unsaturated soil beneath both facilities was requested for each participant [2].

Fig. 1 to Fig. 4 show some of the results presented at NSARS meeting, for four selected radionuclides, covering a wide range of half-lives and distribution coefficients values [3,4].

Although the large dispersitivity observed in the values reported by the participants for maximum flux and the time it occurs, it is possible to identify, in all cases analyzed, a cluster of close results, with variation of less than two orders of magnitude among them. Some clearly outside values presented could be credit to the different input data interpretation, conceptual models, or to the use of numerical codes not appropriate for this specific study case.

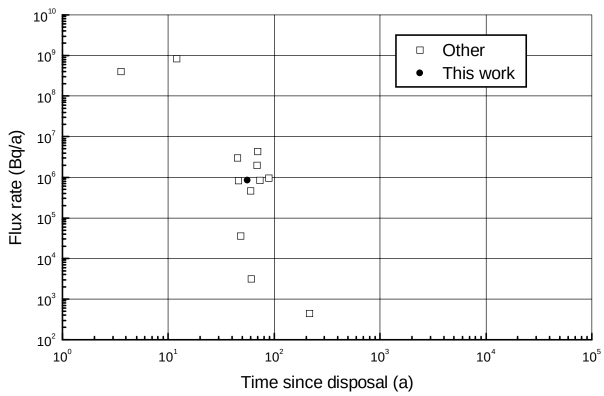


Figure 1. Maximum flux rate to the geosphere and the time it occurs, reported by each participant, considering ^{3}H in the concrete vault.

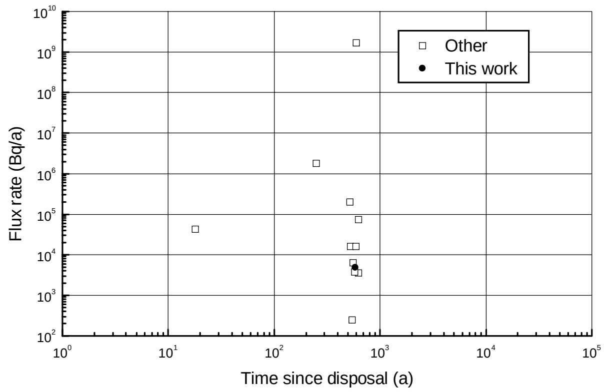


Figure 2. Maximum flux rate to the geosphere and the time it occurs, reported by each participant, considering ^{90}Sr in the concrete vault.

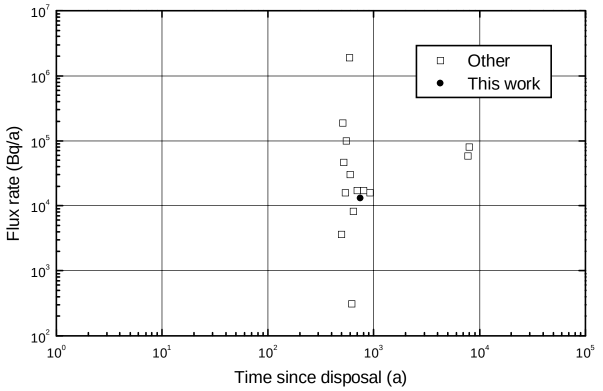
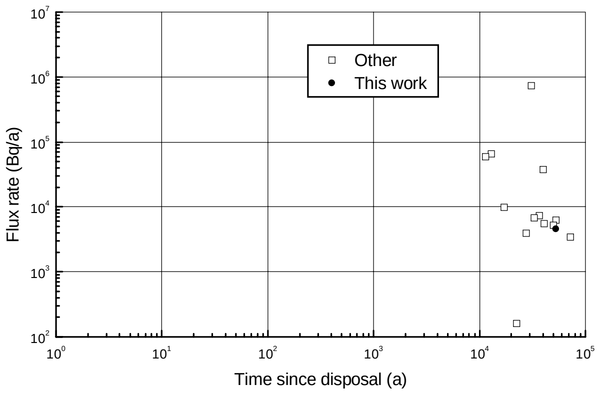


Figure 3. Maximum flux rate to the geosphere and the time it occurs, reported by each participant, considering ^{129}I in the concrete vault.

Figure 4. Maximum flux rate to the geosphere and the time it occurs, reported by each participant, considering ^{230}Th in the concrete vault.



IV. CONCLUSION

The results obtained with this code showed a good agreement with those presented by the majority of the participants. Although one can not say what answers are the *right* or the *wrong* one, this exercise suggests that even a simple code based on the analytical solution of the mass transport equation could be very useful for screening purposes in a safety assessment studies of near-surface waste disposal facilities. The code limitations to accurately solve problems under non-uniform flow conditions needs to be determined in extensive benchmark studies.

REFERENCES

[1] CODELL, R. B. & DUGUID, J. D. Transport of Radionuclides in Groundwater. In: TILL, J. E. & MEYER, H. R., eds. *Radiological Assessment: A Text Book on Environmental Dose Analysis.* Washington, D.C., USNRC, 1983 (ORNL-5968).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY. *NSARS Co-ordinated research programme on The Safety Assessment of Near-Surface Radioactive Waste Disposal Facilities*: *Specification for Test Case 2B.* Vienna, November, 1993.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY. *NSARS 3rd Research Co-ordination Meeting.* Seville, Spain, 25-29 April, 1994 (Technical Reports by the participants).

[4] HIROMOTO, G. *Final Technical Report for Test Case 2B of the IAEA NSARS Programme.* São Paulo, February, 1995.