SIMPLIFIED ANALYTICAL MODEL FOR RADIONUCLIDE TRANSPORT SIMULATION IN THE GEOSPHERE

Goro Hiromoto
Instituto de Pesquisas Energéticas e Nucleares
Comissão Nacional de Energia Nuclear
São Paulo - Brazil

Abstract

In order to evaluate postclosure off-site doses from a low-level radioactive waste disposal facilities, an integrated safety assessment methodology has been under development at Instituto de Pesquisas Energéticas e Nucleares. This paper describes the source-term modelling approach adopted in this system and presents the results obtained in the IAEA NSARS "The Safety Assessment of Near-surface Radioactive Waste Disposal Facilities" programme for models intercomparison studies. The radionuclide released from the waste is calculated using a model based on simple first order kinetics and the transport through porous media bellow the waste is determined using an analytical solution of the mass transport equation. Also, the methodology and results obtained in this work is compared with those reported by others participants of the NSARS programme.

INTRODUCTION

For the present, the low-level radioactive waste generated in Brazil, excepting those generated at Goiania accident, are being temporarily stored in several controlled locations over the country. In order to safely disposal off such wastes, the construction of a national single repository is planned. For the Goiania wastes, a final near-surface repository is being constructed, next to the interim storage site.

In order to evaluate post-closure off-site doses from such waste disposal facilities, an integrated safety assessment methodology is being under development at Instituto de Pesquisas Energéticas e Nucleares.

This paper describes the source-term modelling approach adopted in this system and presents the results obtained in the Test Case 2B of the International Atomic Energy Agency (IAEA) NSARS programme. NSARS is a Co-ordinated Research Programme on The Safety Assessment of Near-Surface Radioactive Waste Disposal Facilities, which has as one of its main objective to improve the reliability of safety assessment methodologies for near-surface radioactive waste disposal facilities, by conducting exercises on model intercomparison/validation.

MODEL DESCRIPTION

A computer code 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. As the output, the code gives the annual radionuclide concentration and flux rate released to the groundwater.

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:

$$\lambda_l = \frac{V}{H(\theta + \rho K_d)}$$

where V is the infiltration rate through the waste, H is the waste depth, θ is assumed to be equal to the unsaturated zone 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]:

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

where

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

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

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

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.

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 characterisation. For results comparison purposes, calculation of maximum flux and concentration in the unsaturated soil beneath both facilities was requested for each participant [2].

Figures 1 to 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, its possible to identify, in all cases analysed, a cluster of close results, with variation of less than two order of magnitude among then. 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.

CONCLUSION

The results obtained with the code developed at IPEN 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 and reliable for safety assessment of near-surface waste disposal facilities.

ACKNOWLEDGEMENTS

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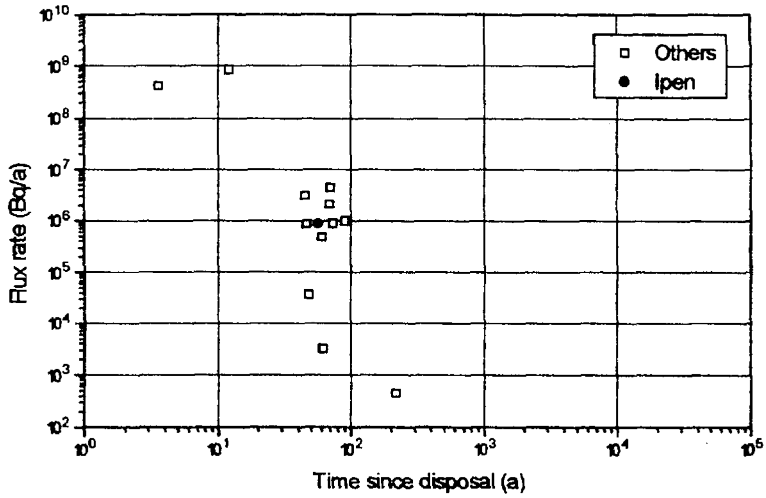

Fig.1 Maximum flux rate to the geosphere and the time it occurs, reported by each participant, considering ^{3}H in the concrete vault.

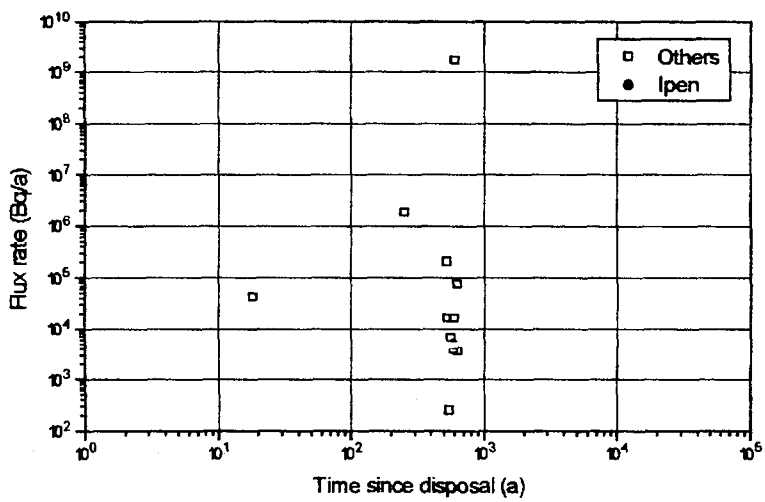

Fig.2 Maximum flux rate to the geosphere and the time it occurs, reported by each participant, considering ^{14}C in the concrete vault.

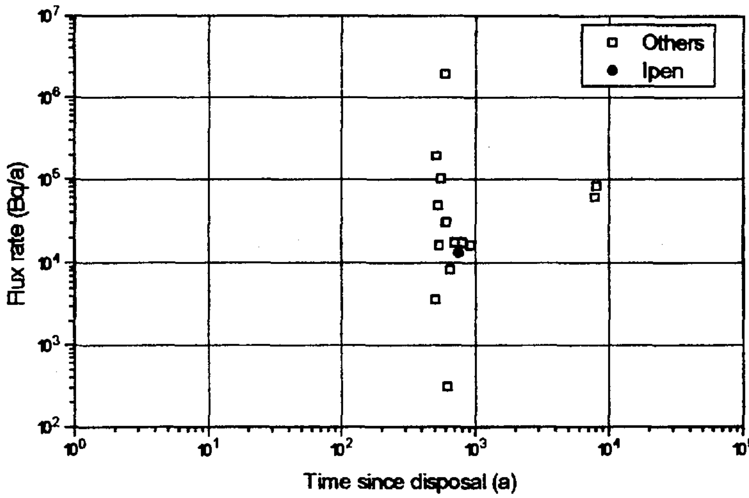

Fig.3 Maximum flux rate to the geosphere and the time it occurs, reported by each participant, considering ^{129}I in the concrete vault.

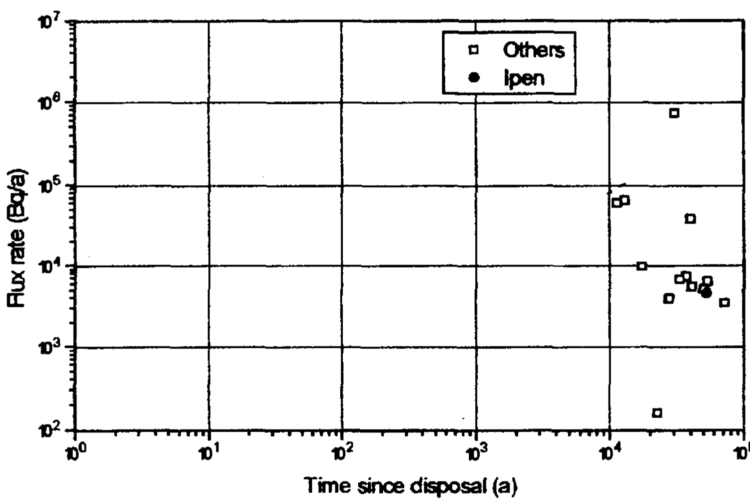

Fig.4 Maximum flux rate to the geosphere and the time it occurs, reported by each participant, considering ^{230}Th in the concrete vault.

RESUMEN

Para avaliar las dosis off-site que pueden ocurrir después de finalizadas las operaciones de almazenamiento de rejeitos radioactivos de baja atividad, se está desenvolviendo, en IPEN, una metodología global de análisis de seguridad radiológica. Este trabajo describe un modelo para ecuacionamento del termo-fonte y muestra los resultados obtenidos en el programa NSARS. La liberación de los radionuclídeos de los rejeitos almazenados és calculada usando un modelo cinético de primera orden y el transporte, atravéz del medio abajo de los rejeitos, és determinado usando un método analítico para resolver la equación de transporte de massas. Los resultados obtenidos utilizando esta metodología también son comparadas con los resultados obtenidos por otros participantes del programa NSARS.