PERFORMANCE ASSESSMENT STUDIES FOR THE LONG-TERM SAFETY EVALUATION OF RADIOACTIVE WASTE DISPOSAL FACILITY.

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Abstract

Especially during the last ten years, a part of Romanian research program “Management of Radioactive Waste and Spent Fuel” was focused mainly on applicative research for the design of near-surface disposal facility, which intends to accommodate the low and intermediate radioactive waste generated from Cernavoda NPP. In this frame, our contribution was at the acquisition of technical data for the characterization of the future disposal facility.

In the present, the project of the disposal facility, located on the Saligny site, near Cernavoda NPP, must be licensed.

As regards to the safe disposal, the location of final disposal, the Saligny site, has been characterized through the five geological formations which contain potential routes for transport of radionuclide released from disposal facility, in the receiving zones(potential receiving zones), into liquid and gaseous phases.

The technical characteristics of the disposal facility were adapted at the Romanian disposal concept using the reference data from IAEA technical report (IAEA,1999). Input parameters which characterized from physical and chemical point of view the disposal system, were partially taken from literature.

The performance assessment studies, which follows the preliminary design development phases and the selection, describes how the source term is affected by the infiltration of water through the disposal facility, degradation process of engineering barriers (reflected in the distribution coefficient values) and solubility limit.

The studies regard the evaluation of the source term, sensitivity and uncertainty analysis provide the information on “how” and “why” were evaluated, following:

- radiological safety assessment of near-surface disposal facility on Saligny site;
- complexity standard assessment of the Engineering Barriers Systems (EBS);
- identification of the elements which must be elaborated for the increase of the disposal safety and the necessity for new technical data for the characterization of the disposal facility.

In the frame of the performance assessment, sensitivity and uncertainty analyses for the Saligny disposal facility have been conducted, consulting associate activity for the three phases corresponding to the disposal:

- operational period;
- institutionalized control;

- post-operative control.

In source term evaluating study and of the sensitivity and uncertainty analysis, we took into consideration radionuclide Co-60, Cs-137, H-3 and C-14, these representing the most relevant radionuclide generated by the operation of the nuclear power plant.

The sensitivity and uncertainty analysis applies to Saligny disposal facility, correlated with special parameters that are influencing the release of radionuclide from repository, was conducted by using a computer code which yielded results allowing the characterization of the disposal facility at the end of the operational period and to eliminate the uncertainties.

1 Introduction

The objective of this paper is to make sensitivity and uncertainty analyses about some parameters that have been effects about gas release. The release it is possible in aqueous and gas phase.

In source term evaluating study and of the sensitivity and uncertainty analyses, we took into consideration radionuclide of Cs-137, H-3 and C-14, these representing the most relevant radionuclide generated by the operation of the nuclear power plant.

Uncertainty is inherent in all performance and safety assessment calculations and regulatory decision-makers need to be informed how uncertainties within the analysis translate into uncertainty in estimates of performance and safety.

Generally uncertainties in assessment are classified as:

- scenario uncertainty;
- model uncertainty;
- parameter uncertainty.

Uncertainty analysis benefits from sensitivity analysis, as results from the latter can be used to reduce the number of parameters for uncertainty analysis. Results from sensitivity analysis may be further used to justify the use of a simple model as a surrogate for a more complex model without loss of important detail, and define priorities for data acquisition. A variety of approaches can be used to identify key sensitivities in the performance and safety assessment analysis, including:

a) calculations in which one parameter related to a single feature or process is varied over a reasonable range of values holding all parameters constant;

b) calculations in which many parameters are varied simultaneously over a reasonable range of values;

c) calculations considering multiple conceptual models.

Sensitivity analysis is done by evaluating effects of perturbations of parameters, models, and scenarios:

- How robust is the disposal system to changes in parameters, models, scenarios?
- improve knowledge on aspects that have greatest impact on model output;
- priorities data collection (directing research towards those site and radwaste properties which contribute most to risk uncertainty;
- which parameters to consider in uncertainty analysis.

A range of different approaches may be used for evaluating uncertainty in performance calculations.

Scenario uncertainty addresses the inability to precisely formulate the future state of the engineered barriers. Scenario uncertainty is generally treated by considering a set of altered evolution scenarios in addition to the expected evolution scenario. To address uncertainty about the future state of repository, altered evolution will be developed.

More specifically, we will develop uncertainty about the future state of engineered barrier:

- hydraulic barrier , use of different about degradation/lifetime;
- concrete degradation (vault/containers/drums). Could consider more conservative assumption about engineering barrier degradation (for example, very simple step function for concrete lifetime can be imposed: from t=0 to t=100 years optimal functioning, for t>100 years complete degradation);
- post-intrusion and intrusion:
    - Gas release as post-intrusion related scenario (as a result of exploratory drilling causing pathways for gas to escape; or as result of cap removal thus providing much shorter travel distance);
    - Exploitation drilling for groundwater: due to core drilling for groundwater exploitation, engineered barriers are short-circuited, and a preferential flow path through the waste zone exists and the unsaturated zone. This results in a (very) small fraction of the waste being leached at a high velocity to the groundwater. Possibly (likely0 the effect on the dose will be very small, but a quantitative evaluation was found justified;
    - Dwelling scenario: two periods of intrusion will be considered soon after end of ICP (Institutionalized control), and one at later time, e.g. after 1000 year to account for effects of radium ingrowths from actinides present in the inventory.
- Climate change: impacts on near field, hydrogeology, biosphere (dryer growing season will require more irrigation possibly groundwater.

  Concept model uncertainty refers to uncertainty about model used to represent a given set of FEP_s (Feature Event Process) and interactions, or choice of models. Uncertainties about the conceptual models relate to uncertainties for models of near field (release or source term models), geosphere models (unsaturated zone and saturated zone), and biosphere models.

  Near field-source term modeling as concerns the leaching pathway. Two approaches have been developed:

    - a generic model with one uniformly distributed source (complete and instantaneous release after ≈30 years) with time-dependent adjective flux;
    - a generic model but with several release functions considered representative for different waste forms (mix of diffusive and adjective processes).

  Near field –gas-release:

    - determination of gas-source (C-14, H-3, Cs-137): quantity/activity available for release via the gas phase;
    - conceptual model for gas transport calculation : we must certify that is a 1D-diffusion equation (or an empirical interpretation of it) a sufficient representation of gas migration processor or is a two-phase representation necessary;
    - we must see what is the effect of using different long-term sorption models and/or different concrete degradation models on the availability of the C-14 source for gas generation and migration. Link between engineered barrier lifetime and maximum gas release rates (100y lifetime versus 500y for concrete EB) and in connection, use of different sorption models for C-14: constant, or kd=f (time).

2 Results and Discussions

For sensitivity analysis, on the basis of the ranking is selected those parameters that have highest impact on dose and consider those for stochastic uncertainty and sensitivity analysis (**Figure 1**).

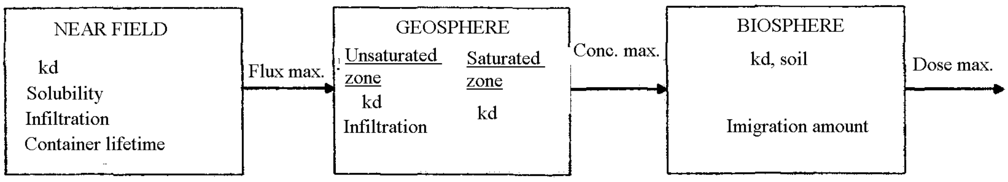

Figure 1: General methodology for selection of parameters calculations

The sensitivity and uncertainty analysis applies to Saligny disposal facility, correlated with special parameters that are influencing the release of radionuclide from repository, was conducted by using a computer code which yielded results allowing the characterization of the disposal facility at the end of the operational period and to eliminate the uncertainties, (Sullivan, 2006).

The activity is assumed to be uniform distributed, so the principal mechanism of release in aqueous phase is diffusion.

The most relevant parameters that were using to make a sensitivity and uncertainty analysis are kd and infiltration rate.

1. The rate infiltration is constant, 2mm / year, with different values for capacity of retaining of concrete;
   - The parameters adequately for afferent region an intact concrete;
   - The parameters adequately for afferent region by degradation concrete;
2. The rate infiltration is variable: for period of 100 year it is considered that is not contact between infiltration water and waste, after 100 year the rate infiltration is 2mm / year and after 300 years, the rate infiltration is 4mm / year;
   - The parameters adequately for afferent region a intact concrete;
   - The parameters adequately for afferent region by degradation concrete.

The values for these parameters are illustrated into **Table 1.**

Table 1: The values for kd and diffusion parameters

	Material Radionuclide	kd [ml/g]		Diffusion [cm2/sec]	
		1 - Top and radier zones	2 – Near field cell	1	2
Intact concrete	**C-14**	2000	2000	1.00E-07	1.00E-06*
	H-3	0	0	1.00E-07	1.00E-06
	Cs-137	2	2	1.00E-07	1.00E-06
	Co-60	100	100	1.00E-07	1.00E-06
	Sr-90	1	1	1.00E-07	1.00E-06
Degradation concrete	**C-14**	500	500	1.00E-07	1.00E-06
	H-3	0	0	1.00E-07	1.00E-06
	Cs-137	0.2	0.2	1.00E-07	1.00E-06
	Co-60	10	10	1.00E-07	1.00E-06
	Sr-90	1	1	1.00E-07	1.00E-06

3 Conclusions

For C-14 radionuclide, it can observed that by raise of infiltration rate, come into a degradation of waste, a containers and engineering barriers who will give a rises about contact between radioactive waste and water infiltration, implicated an more radioactive discharge. The worst cases analysis it was when the rate infiltration is 4mm / year and the distribution coefficient it is for degradation concrete. In this case is obtained a flux by 4.59 x 10^{-6}Bq/sec (**Figure 2:** Calculated flow of C-14 radionuclide into top and radier zones).

For tritium, because it retain is practically zero in geological and in concrete medium, it moves with water infiltration in near field. In this manner, the rate of H-3 is the rate water to the near field. From representative parameters, because H-3 is not retained on the engineering barriers, can be observed a rapid release, the factors who delay the migration are a small permeability of concrete and a small rate of infiltration **(Figure 3:** Calculated flow of H-3 radionuclide into top and radier zones).

For Cs-137, it can observed that the worst case analysis is when the rate of infiltration corresponding to 4mm/y, is obtained a flux by 1.44 x 10^{-2}Bq/sec **(Figure 4:** Calculated flow of Cs-137 radionuclide into top and radier zones.

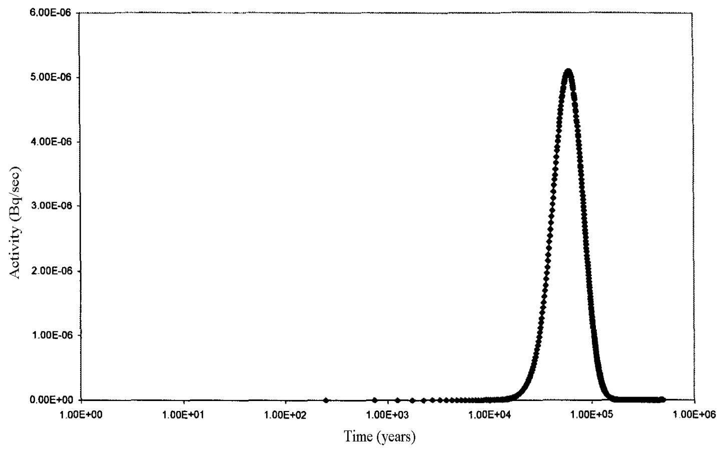

Figure 2: Calculated flow of C-14 radionuclide into top and radier zones

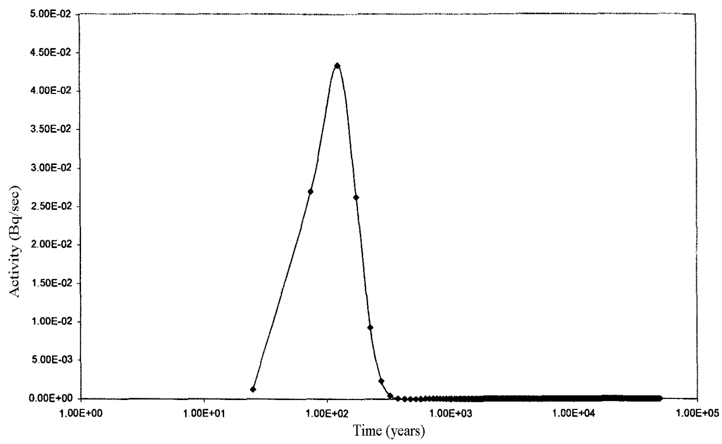

Figure 3: Calculated flow of H-3 radionuclide into top and radier zones

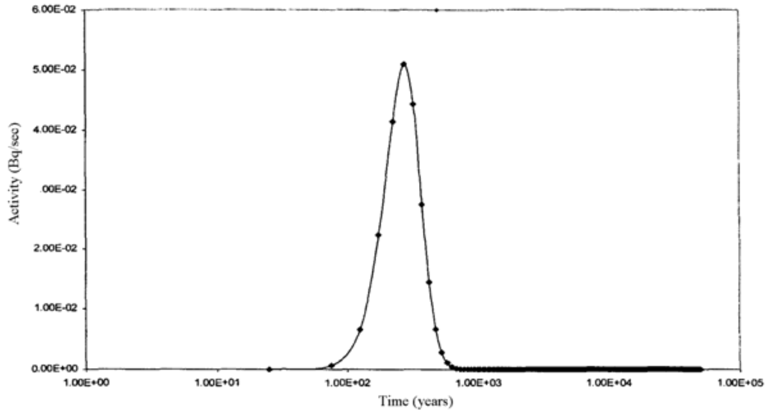

Figure 4: Calculated flow of Cs-137 radionuclide into top and radier zones

References:

IAEA, 1999. Derivation of quantitative acceptance criteria for disposal of radioactive waste to near-surface facilities-development and implementation of an approach. Draft Safety Report Working Documents, Version 3.0, IAEA, Vienna, March 1999;

Sullivan T.M, 2006. Disposal Unit Source Term-Multiple Species Data Input Guide; Environmental Sciences Division Brookhaven National Laboratory Upton, NY.11973.