Deep Geological Disposal Research in Romania

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ABSTRACT: Romania requires a deep geological disposal research for the spent fuel (SF) produced by its nuclear power system based on CANDU reactors. Up to date there is not a final decision concerning the host rock. So, different type rocks are taken into account. Based on two NATO-contracts with the GRS-Braunschweig, namely 1) NATO-LG (ES) 972 750/1988 and 2) NATO-CLG(EST) 976 810/2001, a hypothetical repository has been considered both in granite and salt formations. The key parameter for assessing long-term safety is the radiation exposure to the biosphere. The results of the calculations demonstrate that CANDU SF may be safely disposed.

1 INTRODUCTION

Finding a solution for nuclear waste is a key issue, not only for the protection of the environment but also for future of the nuclear industry. When the first decisions for the emplacement of the existing nuclear power plants will have to be made, the public will require to know the solution for nuclear waste before accepting new nuclear plants. This means that an acceptable solution for the management of nuclear waste is a prerequisite for a renewal of nuclear power.

Deep geological disposal is a mean of safe containment of long-lived radioactive materials as Spent Fuel (SF), discharged from CANDU reactors for instance, for many thousands of years. Deep disposal ensures that any risk from exposure due to accidental intervention or natural disturbance is reduced to a very low level. The main route by which radionuclides in the waste could return to the biosphere is movement in groundwater that may eventually reach the surface to enter the environment. Prior to the construction and operation of a repository, any country would require the proponent, usually the waste management organization, to go through a licensing process with the regulators. This process is aimed at testing the operational and postclosure safety aspects of the concept to demonstrate that the proposal is based on sound scientific knowledge. Such an exercise is often referred to as performance assessment, and the process may involve several iterations as knowledge about a site increases through more detailed site characterization.

Repository postclosure performance assessments attempt to evaluate the radiological safety of a repository after it has been closed and sealed. Different regulators in different countries have their own requirements, but in essence they all require safety performance to be assessed against levels of radiation dose or risk to individuals in the distant future. All

repository concepts are based on the understanding that some radioactivity will be released from the facility at some time in the future and find its way back to human environmental.
The ideal repository would be located in a stable area and would be deep enough to be protected against surface erosion, large climatic changes (such as a new ice age), earthquakes (which are more severe at depth) and human intrusion, of course. It would be located in an impermeable formation and it would guarantee that there would be no release of highly active and dangerous short lived radionuclides during the first thousand years, the time needed for them to decay completely. After that first period, in a longer time scale (over 10 000 years), the geological barrier, will prevent long lived radionuclides leaking significantly into the biosphere, where future generation will live. The time scale issue has particular resonance in relation to: 1) the longevity of the engineered barriers that are intended to keep the radionuclides within the confines of a repository, 2) the rate of radionuclide migration though the rocks surrounding a repository, primarily through transport in groundwater, and 3) the way that one assesses safety for human generation living in the distant future. A useful tool for addressing the first two of these questions is the use of of analog data. In the case of the engineered barriers is essential the study of natural geological systems, the natural analogues, which provide the opportunity to test by observation and measurement, many of the geochemical processes that are expected to influence in a realistic and appropriate way, the predicted reliability of a repository over long periods of geological time. For the second question, that of migration of radionuclides through the surrounding rocks, natural analogs may be also very useful. Natural analogs are occurrences of high concentrations of natural radioactivity similar to those expected in repositories and can make an important input into understanding of repository performance. The third issue, the way that, one judges safety for generations living in the distant future, is usually addressed by producing outcomes for a range of possible situations. In order to get relevant outcomes, mathematical models are utilized to calculate the resultant radiation dose that my arise especially from: 1) the groundwater pathway, in which water slowly move through the repository and may carry away dissolved radionuclides, 2) the gas pathway, in which there could be the release of gases that find their way back to the biosphere, and 3) the human intrusion pathway, in which some future geologic workers may drill into a repository and become exposed.
This is the context in which in Romania has been approached several years ago, in cooperation with GRS-Braunschweig, the research on deep disposal of SF in two geological formations: salt and granite. This was possible due to two NATO contracts, namely: 1) NATO-LG (ES) 972 750/1998 and NATO-CLG(EST) 976 810/2001 of two years for everybody.

2 LONG-TERM SAFETY ASSESSMENT FOR REPOSITORY IN SALT FORMATION

Natural analogues are given by nature. They show the results of natural processes which have lasted sometimes millions of years. Some of these natural processes provide a good example of what can happen in an underground repository and, as such, can bring arguments to overcome the difficult time scale issue. For instance, there are oil and gas fields all over the world covered by sedimentary salt formations, demonstrating the basic feasibility of geological containment and proving that in many cases nature has been able to sustain impermeable conditions for a very long period of time.
The very long half-lives of some of the radionuclides present in SF ensure that some of the waste will remain potentially hazardous, certainly for thousands or millions of years. If we think well, there is nothing so very unusual about this. After all, conventional toxic materials, such as heavy metals, will remain toxic for ever – in effect they have infinite half-lives. However, the question of how waste can be managed over thousands or millions of years raise difficult issues. This is because such time scales are well beyond individual human or

even cultural experience. In spite of this fact, there are a few scientific disciplines, such as geology, archaeology, biology and cosmology which successfully consider and deal with very long time scale, so that the long-term analyses are feasible.
In this frame, a long-term safety assessment of a repository has been performed for SF from CANDU reactors in salt formation. A hypothetical repository site has been considered, using data from European Union Project PAGIS for all parts of the system: near field, overburden, and biosphere. Three scenarios have been taken into account: subrosion as the normal evolution of the salt dome, human intrusion into the cavern representing future human actions, and a combined accident scenario with brine intrusion from the overburden and from undetected brine pockets. Spent fuel elements have been assumed to be disposed of in big storage casks in drifts.

2.1 Subrosion scenario

The scenario assumes that the salt dome is dissolved by groundwater in the cap rock region. The subrosion of the top of the salt dome is balanced by halokinetic uplift, so that the depth of the salt dome top is unchanged. As a consequence, the emplacement field raises continuously and finally reaches the top of the dome. Waste is degraded and dissolved congruently with salt and transported through the aquifers of the overburden to the biosphere. The emplacement sites of the repository are located some 500 m below the top of the present salt dome. The dissolution of the rock salt above the emplacement sites will take about some millions of years. Once this has occurred, the waste will come into direct contact with ground water and the remaining radionuclides will pass into the water. The residual barrier effects of the rock salt and the overburden determine the radiological consequences to the population. The main difficulties encountered in predicting the radiological consequences of such a scenario are related by uncertainties in predicting the subrosion rate, the long-term evolution of the future overburden and ground water movement, and the consumption habits of the future population. Therefore, a simple model, which allows calculation of the radiological consequences, has been proposed for the subrosion scenario [5], [6], and [7]. The difficulties previously mentioned are dealt with by conservative assumptions, which overestimate the consequences of the scenario. A range of parameter values, which are used in local sensitivity analyses, represents the uncertainties in the subrosion rate and in the dilution of the contaminated brine. In order to describe the processes in the future overburden, modeling approaches are selected, which do not require detailed knowledge of future groundwater movement or the structure of the future overburden. Present day consumption habits are used as a basis for intake of radionuclides via drinking water and food chains. The evolution of the scenario can be divided into two time periods. The qualitative and quantitative description of these periods as well as the mathematical modeling is presented in [1] and [5] According to Figure 1, the total volume of the emplacement area, i. e. the volume of dissolution of salt occurs evenly across the whole salt dome, while the dissolution of emplaced canisters surrounded by rock salt can be estimated by the dimensions of the original emplacement area prior to subrosion. All data are given in [1], [3], [5]. Release of radionuclides stretches over a time period which depends on the submission rate and the height of the disposal field. For the range of the subrosion rates of 2.2 mm/y to 0.0005 mm/y and the height of the disposal field of 10 m, the release durations are between 4.5.10E 3 y and 2.10 E 7 y. A release duration of 3.0.10 E 5 y results for the best estimate value of 0.033 mm/y.

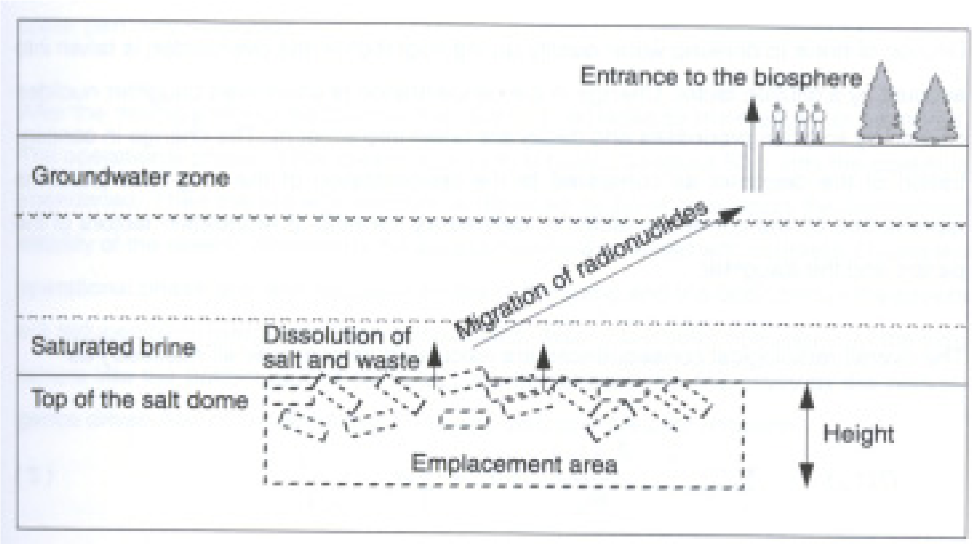

Fig. 1: Subrosion scenario: model of nuclide migration

2.2 Human intrusion scenario

It is assumed that during the solution mining of a storage cavern parts of a 1 000 years old repository with radioactive waste are laid bare. The waste containers in the affected region fall down to the ground of the excavated volume and are buried in the sump of insolubles at the bottom of the cavern. It is likely, that all the containers are then defect and that the corrosion of the waste matrix starts immediately. Due to the continuing excavation process an additional layer of insolubles covers the containers. Thus, the sump is divided into two parts: the bottom part containing insolubles and waste, the upper part only insolubles. After the mining process the brine in the cavern is replaced by the medium to be stored. The operational phase of the cavern is assumed to last for about 50 y until the cavern is abandoned. Then the storage medium is replaced by brine to support the mechanical stability of the cavern. Afterwards the access borehole is sealed with concrete.

2.3 Combined accident scenario

The repository is modeled according to a hypothetical repository similar to that described in ref. [9], [10]. In the following, a short overview is given of the general modeling procedure. The source term for the release of radionuclides from spent fuel elements is assumed to be the same for spent CANDU and LWR fuel elements, but some data are assumed to be different. The modeling of the source term is presented in detail in [1]. Finally, some relevant input data for the near field are compiled. The modeling and input data for the geosphere and biosphere are taken from refs. [10] and

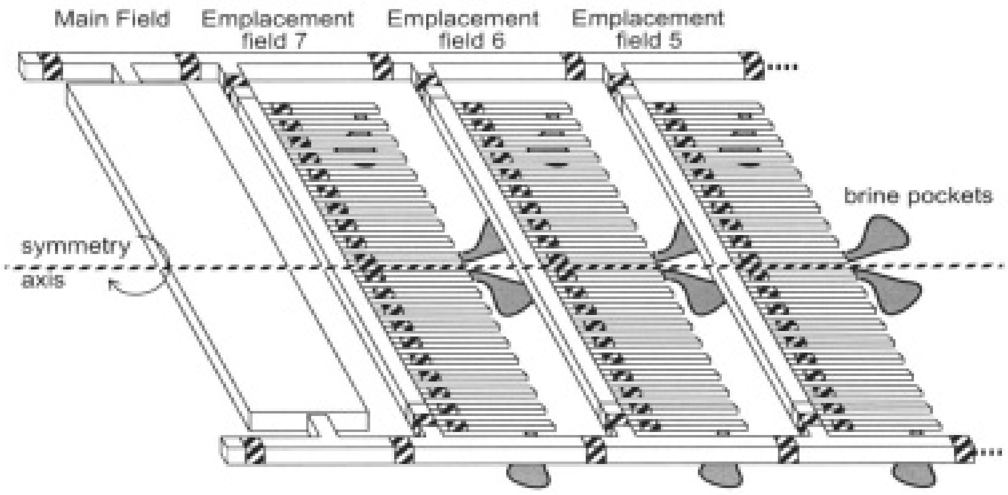

Fig. 2. Part of the section system of the near field.

The entire repository consists of 7 emplacement field [11] without changes. The section structure of the near field is schematically shown in Figure 2. The main field with infrastructure reason is followed by 7 emplacement fields, each with 20 emplacement drifts for containers with spent fuel. It is assumed that in each emplacement field 4 brine pockets occur, 2 pockets connected to the drifts at the end of the field and 2 pockets connected to the drifts in the center of the field. The infrastructure region of the main field is modeled as a drift system with an additional open void. The drifts are modeled as porous media with a flow resistance according to the permeability, and the additional voids are assumed to be accessible to brine but not contributing to the flow resistance. Due to the symmetry as shown by the symmetry axis, a tree like structure of the near field can be modeled.

2.4 Input data

All the input data are based on those given in ref.. [12]. The data for the geosphere and the biosphere are taken over without changes, because the radiation exposures are calculated only as a tool to compare the results. The intention is, to compare the influence of different fuel types, and this influence is only given for the near field. For the near field most of the data are taken from ref. [9],[10]. First, the data that are common to all scenarios are presented. Among these are: temperature data, container and fuel design data, the geometrical data of the repository, waste inventories, solubilization data and element specific solubilities, and dose conversion factors. Additional data concerning the geometry of the repository (i. e., the section systems of the repositories and the specific geometrical section data) and the sorption data are presented in Ref. [8], [12]. Finally, specific data for every scenario are presented. The Subrosion scenario requires knowledge of the following data: time of begin of release from the repository, salt mass in the emplacement area, and concentration of salt in drinking water. For the Human intrusion scenario, the following parameters are required: geometrical data (radius and height of the cavern and also of the cavity, and final pore volume of the cavity), maximum value of brine pressure, initial and reference value for the porosity of the sump, and time of spontaneous fill-up of the cavity. Two more specific data are necessary for a complete description of the Combined scenario. These are: time of brine intrusion from the anhydrite vein (unlimited brine intrusion), and time of brine intrusion from the brine pockets (limited brine intrusion).

2.5 Results

All results have been obtained with the german computer code EMOS 5.2 [3].
The consequences of the three scenarios are calculated in two ways: by best estimate calculations and local sensitivity analyses. Best estimate calculations are deterministic calculations with best estimate values for the input parameters and some conservative assumptions in the modeling. Results are presented as doses to individuals for each scenario, neglecting the probability of occurrence. Local sensitivity analyses are best estimate calculations with one single parameter being varied over its assumed 3-σ range to study the behavior of the system. Results of local sensitivity analyses are obtained as released nuclide masses or as individual doses for each scenario as a function of the considered input parameter, but for economy reasons will be not inserted in this paper.

2.5.1 Results for subrosion scenario

As known the subrosion rate determines the time of release of radionuclides and the radionuclide activity, too. In the best estimate case a subrosion rate of 0.033 mm/y is assumed which results in a time of release of 1.5_10 7 y. Time of release means the onset of a radionuclide flux into the groundwater. With this value the resulting maximum of the radiation exposure from CANDU fuel is 3.7_10 -4 Sv/y (and in the LWR case 1.5_10 -4 Sv/y). The main contributor to dose is U-234 (Ra-226) in both cases. Although these radiation exposures are rather high compared to the results of the combined accident scenario the conservative assumptions in the model have to be kept in mind. The results of the best estimate calculations are listed in Table 1.

Table 1. Subrosion scenario: Maximum total doses [Sv/y] for CANDU (and LWR) fuel as function of subrosion rate and concentration of salt and concentration of salt in drinking water.

Subrosion rate [mm/y]	Salt concentration [mg/l] 1		31 (best estimate)		1000	
	CANDU	LWR	CANDU	LWR	CANDU	LWR
2.2	$1.7\cdot10^{-5}$	$4.8\cdot10^{-5}$	$5.3\cdot10^{-4}$	$1.5\cdot10^{-3}$	$1.7\cdot10^{-2}$	$4.8\cdot10^{-2}$
0.5	$1.4\cdot10^{-5}$	$1.9\cdot10^{-5}$	$4.4\cdot10^{-4}$	$5.9\cdot10^{-4}$	$1.4\cdot10^{-2}$	$1.9\cdot10^{-2}$
0.033 (b. e.)	$1.2\cdot10^{-5}$	$4.7\cdot10^{-6}$	$3.7\cdot10^{-4}$	$1.5\cdot10^{-4}$	$1.2\cdot10^{-2}$	$4.7\cdot10^{-3}$
0.005	$1.2\cdot10^{-5}$	$4.1\cdot10^{-6}$	$3.6\cdot10^{-4}$	$1.3\cdot10^{-4}$	$1.2\cdot10^{-2}$	$4.1\cdot10^{-3}$
0.0005	$1.0\cdot10^{-5}$	$3.5\cdot10^{-6}$	$3.1\cdot10^{-4}$	$1.1\cdot10^{-4}$	$1.0\cdot10^{-2}$	$3.5\cdot10^{-3}$

2.5.2 Results for human intrusion scenario

In this scenario, the temperatures in the neighborhood of the waste containers after 1000 y can only be roughly estimated. By these preliminary calculations for CANDU fuel, three different sets of temperature have been chosen for the cavern sections. These calculations have been performed to demonstrate the negligible influence of a detailed modeling of the temperature fields. The results are listed in Table 2. It turned out that, due to the generally low temperatures of the emplacement field after 1 000 y, it is of minor importance which of the three options is used. Thus, in order to be conservative, the calculations are performed with the first set of temperatures, which yields the highest radiological consequences. The temporal evolution of the scenario starts at t =0 y by leaching the waste matrices, because containers are assumed to be defect from the beginning of the scenario. After 50 y, when the cavern is sealed, the porosity of the sump has gained a value of 0.365 (0.217 for LWR fuel) due to convergence. After that time, the brine pressure rises within less than 0.5 y from the hydrostatic level of 10 MPa to the given maximum value of 15 MPa. At this pressure it is assumed, that the sealing of the cavern cracks and that the contaminated brine start to pass into the geosphere. The increase of brine pressure slows down the convergence, which reduces the brine flow.

Table 2: Preliminary tests on the influence of section temperatures to radiological consequences. CANDU fuel.

Table 4.

No.	Temperature data	Time of cavern closure [y]	Time of dose maximum [y]	Maximum dose [Sv/y]
1.	- sump temperature corresponds to a field with highest temperatures - cavern has rock temperature at the reference level (i. e., no height dependence)	$8.89 \cdot 10^{5}$	$4.94 \cdot 10^{5}$	$6.80 \cdot 10^{-5}$
2.	- sump temperature corresponds to a field with highest temperatures - cavern has rock temperature at the centre of the cavity	$9.02 \cdot 10^{5}$	$5.05 \cdot 10^{5}$	$6.46 \cdot 10^{-5}$
3.	- every section has rock temperature at the corresponding depth	$9.03 \cdot 10^{5}$	$5.05 \cdot 10^{5}$	$6.47 \cdot 10^{-5}$

The corresponding temporal evolution of the annual radiation exposure is shown in Figures 3. With CANDU fuel, the annual radiation exposure attains a maximum of 6._10 -5 Sv/y at t =494 000 years. The main contributions to the radiation exposure are from Np-237, followed by U-233, I-129 and Tc-99. Se-79 dominates the radiation exposure at early times.

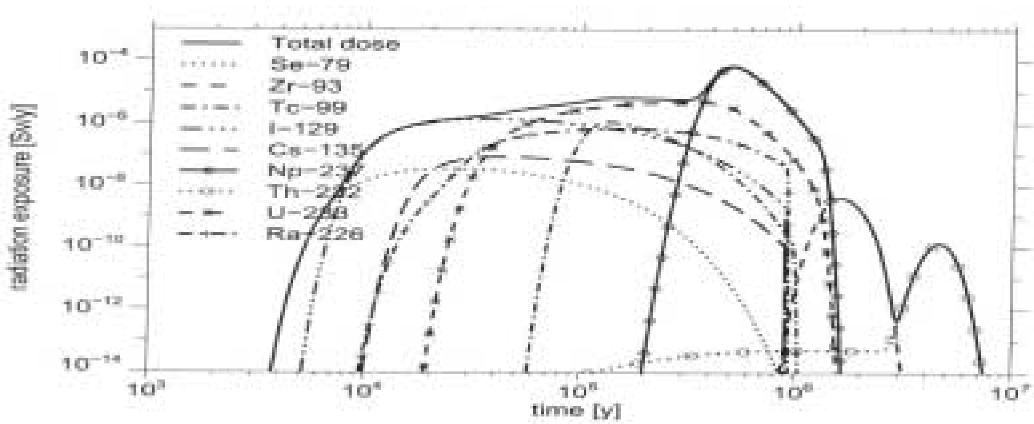

Fig. 3: Human intrusion scenario (CANDU Fuel): Temporal evolution of the radiation exposure in the reference case.

2.5.3 Results for combined accident scenario

The results are presented for CANDU fuel.. The best estimate values of the input parameters are also used.The time history of some relevant segments and the amounts of water entering the entire repository are listed in [1]. It can be seen that all the emplacement fields contribute to the radionuclide release, because brine from the brine pockets keeps most of the emplacement drifts open. Thus the intruding brine from the main anhydrite can flow from the central field into the emplacement areas. Only those emplacement drifts that are not connected to brine pockets do not contribute to the release of radionuclides. Because there is no brine to delay the convergence process, these emplacement drifts are closed by convergence and the entering brine remains trapped inside the drifts. Almost 15 845 m 3 of water enters into the repository, 11 347 m 3 intruding from the geo-sphere and 4 498 m 3 from the brine pockets. About 11 857 m 3 of contaminated water is squeezed out from the repository over the time period of 1 million years, which is a little bit more than the total inflow from the geosphere. By convergence, almost 80% of the volume of the brine pockets is squeezed out into the emplacement drifts. The main contributions to the total dose are by I-129, followed by Ra-225, Np-237, and U-233. The temporal evolutions of the radiation exposures for several of the most important nuclides are shown in Figure 4 The total dose rises to about 9_10 -6 Sv/y at 440 000 y, and is more than one order of magnitude below the limit of the radiation protection law.

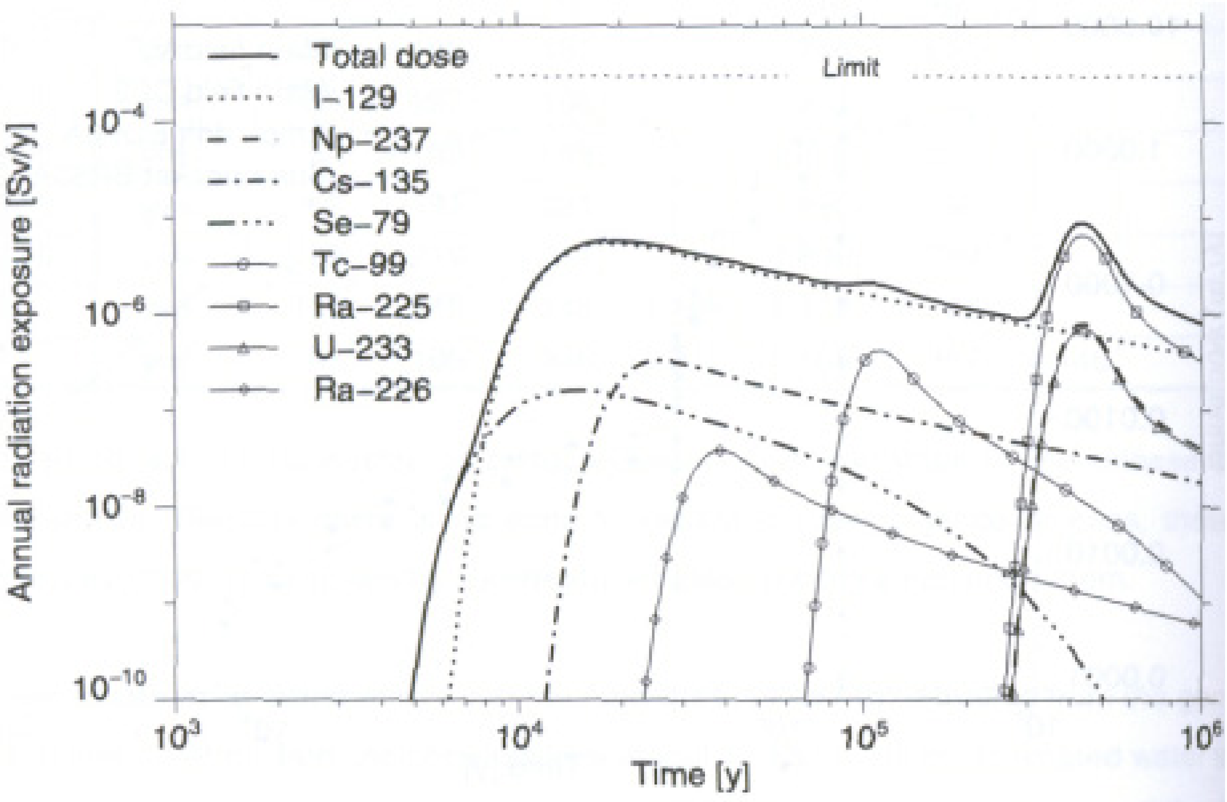

Fig. 4. Combined accident scenario (CANDU fuel): Temporal evolution of the radiation exposure in the reference case

3 LONG-TERM SAFETY ASSESSMENT FOR REPOSITORY IN GRANITE FORMATION

The aim of the present research is to perform a long-term safety analysis for direct disposal of spent nuclear fuel from power plants of CANDU type. CANDU reactors use natural uranium as nuclear fuel. The study was jointly performed by a Romanian institute and a German company. Thus, the input data for the calculations are taken from romanian national data bases. Disposal is assumed to take place in a granite formation. The long-term safety of such a repository , but for LWR SF, has been investigated in detail as part of a recent project of the European Commission, called Spent fuel disposal Performance Assessment (SPA

project) [1]. Results of the German participant regarding spent LWR fuel have been published in detail separately [11]. The latter report is the basis for the actual investigations.
After contact with groundwater, the radionuclides disposed of in the repository are assumed to be released from the spent fuel elements and to be transported through the repository system by diffusive and advective flow. Other release paths are not dealt with in this study. After transport of the radionuclides to the biosphere a radiation exposure to man occurs. Thus, the consequences of a potential release of radionuclides are mainly discussed in terms of annual radiation exposures, i.e. effective doses.
The consequences are calculated by a deterministic method applying the computer codes GRAPOS, CHETMAD, and EXMAS of the computer code package EMOS [3], all of them in the version 1.01. Release from the near field is calculated by GRAPOS, transport in the geosphere by CHETMAD, and the radiation exposure in the biosphere by EXMAS. Uncertainties of the input parameters are treated by local parameter variations. The repositories for spent CANDU or LWR fuel elements are assumed to be different mainly in the data of the source term for spent fuel and the size of the underground facilities. Differences in the amount of waste of both countries are also taken into account. A common model of the source term is applied for both types of fuel.
The general concept for disposal of spent fuel elements in granite is similar for CANDU and LWR. The containers with spent fuel are placed central in cylindrical emplacement boreholes, and are surrounded by cylindrical buffer material. The buffer is assumed to enclose the containers entirely. Thus, there is no advective flow through the buffer and the release of radionuclides in this part of the repository system is only by diffusion. Flow of water around the buffer is assumed to be possible through the excavation damaged zone (EDZ). If a connection exists between the EDZ and a major water conducting fault in the granite body, then radionuclides can be transported advectively from the EDZ towards the biosphere. This is considered the reference scenario and it is the single one taken into account in this study.

3.1 Results for reference scenario

In the following, results of the calculations for the reference case are presented for CANDU SF [13].
The maximum release rates and concentrations from the geosphere into the biosphere are given in Figure 5 .

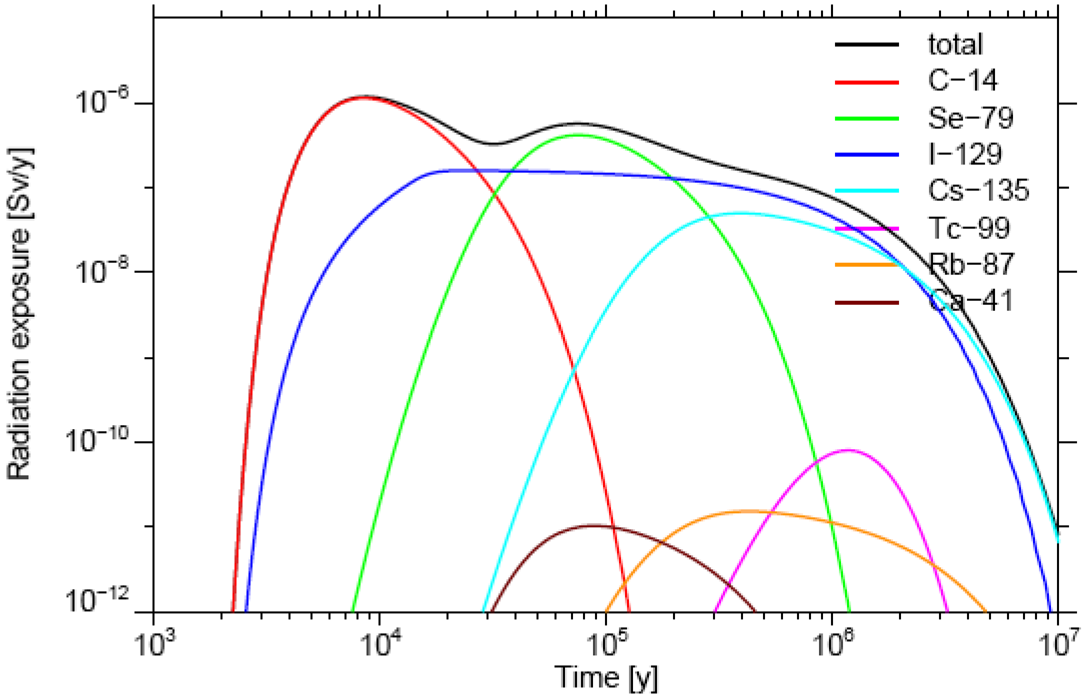

Fig. 5. Release rates of relevant radionuclides from the geosphere into the biosphere

4 SUMMARY AND CONCLUSION

Performance assessmets for a hypothetical repository for CANDU SF in salt formation have been performed in terms of effective doses for three scenarios: subrosion, human intrusion and combined accident.

- For the subrosion scenario with best estimate values of the input parameters, the radiation exposures are at the radiation protection limit of 3_10 -4 Sv/y. A maximum radiation exposure of 3.7_10 -4 Sv/y has been calculated for CANDU fuel. The main contributor to radiation exposure is U-234.
- In the human intrusion scenario with best estimate values of input parameters, the maximum radiation exposure for CANDU fuel is 6.8.10-5 Sv/y and is mainly caused by Np-237. Thus, the consequences of the human intrusion scenario are acceptable, especially because its probability of occurrence is low.
- In the combined accident scenario an intrusion of brine from the overburden into the repository via the main anhydrite is considered in combination with a limited brine intrusion from brine pockets which are located in the neighborhood of the emplacement galleries. Applying best estimate values of the input parameters, the maximum radiation exposures for this scenario are 9_10 -6 Sv/y for CANDU The most relevant nuclides are I-129 and Ra-225 for CANDU fuel.

In case of granite formation only reference scenario has been taken into account for a hypothetical repositoy for CANDU SF.

- Sensitivity analyses must be performed for spent fuel safety or performance assessment in order to predict the final consequences of the uncertainties related to poor or insufficiently accurate data.
- Four input parameters (the water flow through the EDZ, the bentonite buffer thickness, the geosphere water flow rate and the biosphere dilution factor) were identified, whose variations may lead to important consequences in the safety or performance indicators (at least one order of magnitude.

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