EUROPEAN COMMISSION

Community research

PAMINA
Performance Assessment Methodologies in Application to Guide the Development of the Safety Case

(Contract Number: FP6-036404)

SAFETY INDICATORS AND PERFORMANCE INDICATORS
DELIVERABLE (D-N°:3.4.2)

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Date of issue of this report : **30/09/09**

Start date of project : **01/10/2006**

Duration : 36 Months

Project co-funded by the European Commission under the Euratom Research and Training Programme on Nuclear Energy within the Sixth Framework Programme (2002-2006)		
Dissemination Level		
PU	Public	X
RE	Restricted to a group specified by the partners of the [PAMINA] project	
CO	Confidential, only for partners of the [PAMINA] project	

Foreword

The work presented in this report was developed within the Integrated Project PAMINA: **P**erformance **A**ssessment **M**ethodologies **IN** **A**pplication to Guide the Development of the Safety Case. This project is part of the Sixth Framework Programme of the European Commission. It brings together 25 organisations from ten European countries and one EC Joint Research Centre in order to improve and harmonise methodologies and tools for demonstrating the safety of deep geological disposal of long-lived radioactive waste for different waste types, repository designs and geological environments. The results will be of interest to national waste management organisations, regulators and lay stakeholders.

The work is organised in four Research and Technology Development Components (RTDCs) and one additional component dealing with knowledge management and dissemination of knowledge:

In RTDC 1 the aim is to evaluate the state of the art of methodologies and approaches needed for assessing the safety of deep geological disposal, on the basis of comprehensive review of international practice. This work includes the identification of any deficiencies in methods and tools.

In RTDC 2 the aim is to establish a framework and methodology for the treatment of uncertainty during PA and safety case development. Guidance on, and examples of, good practice will be provided on the communication and treatment of different types of uncertainty, spatial variability, the development of probabilistic safety assessment tools, and techniques for sensitivity and uncertainty analysis.

In RTDC 3 the aim is to develop methodologies and tools for integrated PA for various geological disposal concepts. This work includes the development of PA scenarios, of the PA approach to gas migration processes, of the PA approach to radionuclide source term modelling, and of safety and performance indicators.

In RTDC 4 the aim is to conduct several benchmark exercises on specific processes, in which quantitative comparisons are made between approaches that rely on simplifying assumptions and models, and those that rely on complex models that take into account a more complete process conceptualization in space and time.

The work presented in this report was performed in the scope of RTDC 3.

All PAMINA reports can be downloaded from http://www.ip-pamina.eu.

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1. Introduction

The disposal of radioactive waste in deep geological formations implies a potential hazard to man and the environment. Therefore the most important task for the process of siting and designing a disposal system is to assure for a very long time period that the disposed waste causes no harm for human health and the environment.

A safety case is the synthesis of evidence, analyses and arguments that quantify and substantiate a claim that the repository will be safe after closure and beyond the time when active control of the facility can be relied on (NEA 2004). An important part of every safety case is the evaluation of the long-term radiological consequences for a variety of relevant scenarios that seem possible or, at least, cannot be excluded. The primary outcomes of such calculations are radionuclide activity fluxes at different locations in the system, which themselves have no direct relevance for safety. To allow an assessment of long-term safety, it is necessary to calculate from the fluxes at least one safety-related measure and to compare it with a suitable reference value. Such magnitudes are called safety indicators.

Most national regulations give safety criteria in terms of dose and/or risk, which are evaluated for a range of evolution scenarios for the disposal system using mathematical analyses. Dose calculations include complex exposure pathways depending on biological characteristics of different species as well as on human behaviour. There is a high level of uncertainty about the assumptions that are used when calculating doses to humans. Besides the near-surface processes, which are difficult to predict over long time-scales, in particular the usual assumption that the biosphere properties remain unchanged for the long time period considered in a safety case is highly questionable. Consequently, the use of dose or risk as the only safety indicator has a clear disadvantage due to the uncertainty surrounding their estimation (IAEA 2005).

The robustness of the safety case can be strengthened by the use of multiple lines of evidence leading to complementary safety arguments that can compensate for shortcomings in any single argument. One important type of evidence and arguments in support of a safety case is the use of safety indicators complementary to dose and risk.

Complementary safety indicators can avoid, to some extent, the uncertainties of doses and risks. In contrast to near-surface and biosphere properties, the possible evolutions of a well-chosen host formation can be predicted with reasonable confidence over much longer time scales. Hence, there is a trend in some recent safety cases towards evaluating other safety indicators, in addition to dose and risk, such as radiotoxicity fluxes out of the geosphere, which do not rely on assumptions about human behaviour and can support the safety statement and increase the robustness of the safety case.

Safety indicators provide statements about the overall safety of a repository system. Additionally it can be valuable to investigate the functioning of the repository system and its components on a more technical level by calculating quantities that describe the effectiveness of individual barriers or parts of the system. Such quantities are called performance indicators. Typical performance indicators are radionuclide concentrations and fluxes in or between different parts of the system. They provide a good means for understanding and communicating the functioning of the system and can support the safety case in an illustrative manner.

Because of the importance of safety and performance indicators this subject has become one of the main topics of PAMINA RTDC 3 and was included there as working package (WP 3.4). The three principal objectives of WP 3.4 were to

- achieve a common understanding for the terms safety indicators and performance indicators,
- test appropriate safety indicators and performance indicators for the three host rock types (clay, granite, and rock salt) considered within the EU for deep geological repositories, and
- determine adequate reference values for the considered safety indicators.

The first objective was discussed intensively within the first phase of PAMINA. The results of this discussion are summarized in Chapter 2. It reflects the common understanding of the terms safety indicator, performance indicator and important related terms. At the end, the general purpose of these indicators for a safety case is summarized in a schematic illustration.

For the second main objective the participating organisations performed calculations for three host rock types: clay, rock salt and granite. Chapter 3 describes the disposal concepts, the underlying assumptions and the relevant parameters used by the participating organisations for the calculations of the applied indicators.

Chapter 4 deals with the safety indicators and the corresponding reference values used in WP 3.4. Because of the importance of the reference values (third objective of WP 3.4) chapter 5 gives a look in more detail on the work performed on reference values. Finally, chapter 6 summarizes the results of the safety indicator calculations.

Chapters 7 and 8 describe the applied performance indicators, the underlying compartment structures and the results of the calculations.

The last chapter gives a summary of the lessons learnt from the work performed in WP 3.4.

This report is the final report of all participating organisations of PAMINA WP 3.4. It summarizes the results obtained by the participating organisations. Each organisation prepared an individual report, which can be downloaded from http://www.ip-pamina.eu.

2. General concepts

The use of complementary safety indicators for assessing the overall safety of a repository as well as performance indicators for demonstrating the functioning of the system has been widely discussed in international fora and projects, for instance in IAEA (2005). In the SPIN project (Becker et al. 2003) it has become clear, that the terms “safety indicator” and “performance indicator” are not used in exactly the same sense throughout the literature. Therefore, specific definitions were established for the purpose of SPIN. In view of newer developments, however, it seems necessary to refine these definitions.

The definitions that were used in PAMINA work package 3.4 in order to create a common basis are given in this chapter. These definitions are neither intended to replace any existing definitions nor to anticipate the results of any discussions on the subject. The work performed in WP 3.4 is intended to contribute to these discussions.

Based on the derived definitions the role of safety and performance indicators in a safety case is summarized in chapter 2.4. A description in more detail can be found in Becker and Wolf (2008).

2.1 Safety indicators

As stated in the introduction the assessment of the long-term safety by means of model calculations requires calculable quantities that are related to safety and that can be compared with reference values. For the purpose of WP 3.4 the following definition is applied:

> A safety indicator is a quantity, calculable by means of suitable models, that provides a measure for the total system performance with respect to a specific safety aspect, in comparison with a reference value quantifying a global or local level that can be proven, or is at least commonly considered, to be safe.

A safety indicator should give an indication of whether a repository can be considered safe regarding some safety aspect, e.g. human health or groundwater quality.

> A safety aspect is a specific subarea of the total field of long-term safety that can be addressed independently of others.

According to the results of a numerical assessment a repository can be safe with respect to one aspect but unsafe with respect to another.

A safety statement related to a defined safety aspect requires a numerical measure as well as a reference value defining a safe level. A safety indicator without a reference value cannot be used to assess the long-term safety of a repository system. It is recommended not to speak of a safety indicator without a reference value specified.

When determining a reference value for a specific safety indicator it is essential to take account of the underlying safety aspect. The same numerical measure for repository safety, even when calculated in exactly the same way, can yield different and independent safety statements, if referred to different safety aspects and combined with the appropriate reference values.

Reference values can be valid globally like the concentration of radiotoxicity in drinking water that is harmless for human health. Other reference values have a very local character and are only valid in a specific environment, e.g. natural radiotoxicity flux in groundwater (chapter 5).

2.2 Performance indicators

Safety indicators are a good means for assessing the level of safety of the total system, but they do not provide detailed information about how the system works and how the level of safety is reached. That is the objective of a performance indicator.

> A performance indicator is a quantity, calculable by means of appropriate models, that provides a measure for the performance of a system component, several components or the whole system in comparison with each other.

Performance indicators do not need reference values or technical criteria.

It was shown in SPIN that the applied performance indicators for a repository in granite are a good means to visualise the functioning of the system and to help understanding the coaction and interaction of its components.

Performance indicators are typically activities, concentrations or fluxes of radionuclides in or between specific parts of the repository system, or other descriptive measures that demonstrate specific properties of the system.

The general idea of the concept of performance indicators is to look in detail at the transport processes at specifically relevant locations inside the repository system. Interesting locations can be, e.g., the interfaces between parts of the system that fulfil different tasks. Comparing the indicators calculated for different locations is often very illustrative for demonstrating the functioning of the system. In view of this purpose, the division into part-systems has to be done carefully. Following the SPIN terminology, these part-systems are called compartments. Compartments can be natural or mined subsystems like the geosphere or the mine building, engineered components like canisters or barriers, or even physically independent phases in specific regions, like the canister water or precipitate. There is no need to have a non-overlapping compartment structure. It often makes sense to consider compartments that contain others. An example is the “canister” compartment, which contains the three compartments waste matrix, canister water and precipitate. When designing the compartment structure for a specific repository system, one has to distinguish whether the processes in the compartments or between the compartments are to be illustrated. Normally, these two purposes lead to different compartment structures.

2.3 Safety function and safety function indicators

In SPIN, the term "safety function" was used for three basic tasks of a repository system: physical containment, delay and decay, dispersion and dilution. For the purpose of WP 3.4 the following definition is applied:

> A safety function is a functionality the repository system should provide as a whole, in order to prohibit or reduce detrimental effects.

In the concept developed and applied by SKB (2006), a safety function is a role through which a repository component contributes to safety such as the isolation potential of a container or the retardation capability of a barrier. In this concept a safety function indicator is a measurable or calculable quantity through which a safety function can be quantitatively evaluated. Each safety function indicator is associated with a specific criterion and can be assessed independently of others.

The concept applied by SKB uses the term safety function (and the corresponding term safety function indicator) on a very detailed component structure, whereas the definition given above defines only a functionality of the whole system as a safety function. These definitions are not that different, but they can lead to some confusion in the discussion on indicators for a safety case. To avoid confusion the term "safety function indicator" is not used in this report.

It is recommended that the terms safety function and safety function indicator are only used, if it is clearly defined what kind of safety functions are considered in the safety case and what the differences are between safety indicators, performance indicators and safety function indicators.

2.4 The use of safety and performance indicators in a safety case

As stated in chapter 2.1 every safety indicator that is used in the safety case must provide a safety statement that is related to a previously defined safety aspect. These safety aspects are based on at least one of the two basic safety concerns "protection of human life" and "protection of environment" (according to IAEA 2006). A safety statement could for example be that all biological effects to a human individual, i.e. the incorporation of radionuclides released from a repository by humans via different exposition paths, remain so small that they have no or only a very small impact on human health. The corresponding safety aspect is the human health. For a complete safety statement as defined in chapter 2.1 a numerical measure is required, by which the biological effects due to incorporation of radionuclides can be calculated, as well as a reference value in order to define a safe level. A common measure for the exemplary safety statement is the effective dose rate. In the case of the effective dose rate, the national legal limit can be used as reference value (chapter 4.1).

In order to use a safety indicator in a safety case, it is necessary to calculate the outcome of the considered numerical measure (e.g., the effective dose rate) by means of a PA model for the corresponding repository system. By comparing the results of the PA model with the reference value of the safety indicator, the determination of the safety indicator is complete and can be added to the safety case (Figure 2.1).

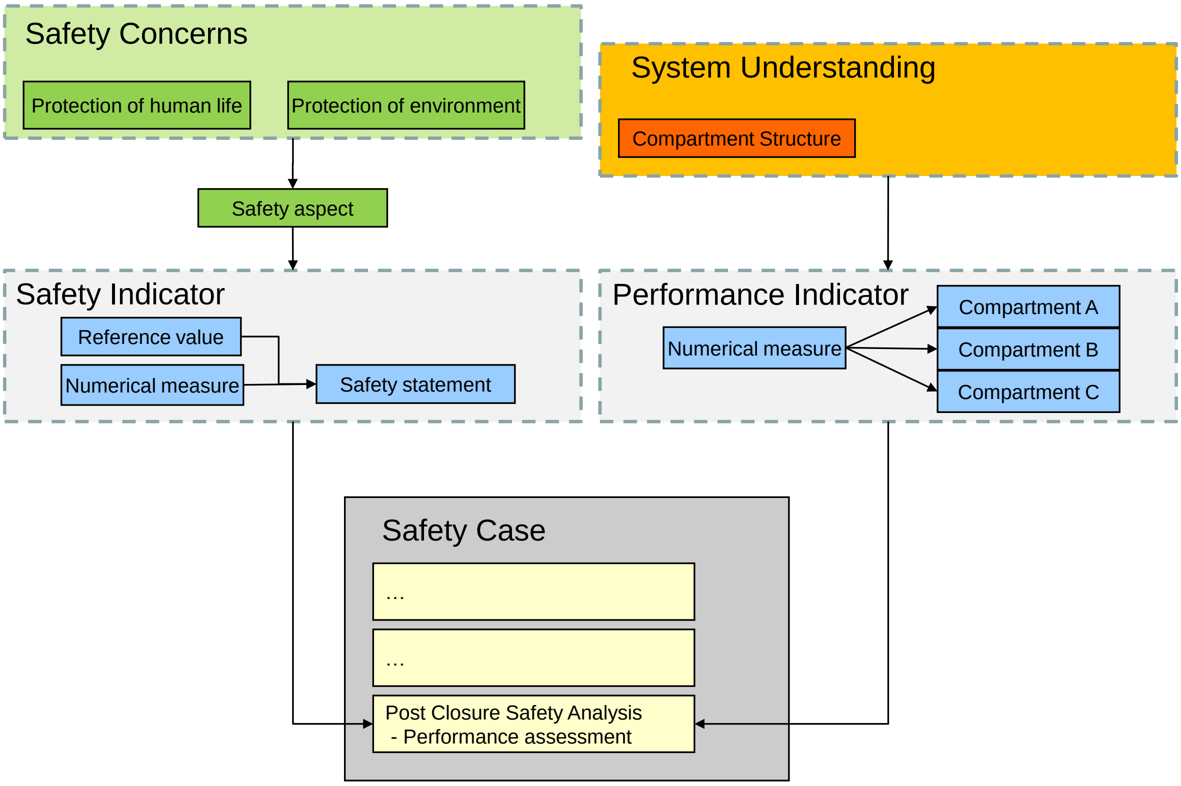

Figure 2.1 General concept for the use of safety and performance indicators for supporting the Safety Case

In contrast to safety indicators there can be a close interaction between the PA model and the definition and calculation of performance indicators, especially between the compartment structure and the repository design. For every performance indicator a compartment structure has to be defined. The chosen compartment structure depends on several conditions, for instance the host rock or the type of quantity (concentration, flux) that is calculated. If several performance indicators are used within one study, the compartment structures should be identical, or at least as similar as possible, in order to allow a comparison of the results.

Since performance indicators are not based on safety statements their use is more flexible. Performance indicators provide information about how the system works and how the level of safety is reached. Such information is of high value for the safety case. For the experts it is essential to understand how the different barriers act together and where the radionuclides are mainly retained. Performance indicators are useful for optimisation of the disposal system, comparison of different options, improving the understanding of the role played by different system components and communication of these aspects, both to experts and the general public.

For communication with licensing authorities as well as with the general public it can be helpful to demonstrate the functioning of the system in an illustrative and understandable way. Such demonstrations can improve the confidence in the safety case.

At the end not every applied performance indicator is added to the safety case, but most of them give valuable arguments for increasing the confidence in the safety of a repository system (Figure 2.1).

3. Repository systems

For PAMINA WP 3.4 six repository systems were defined:

- three systems in clay formations (ENRESA, NRG, SCK·CEN),
- two systems in rock salt formations (GRS, NRG),
- one system in a granite formation (NRI).

3.1 Repositories in clay formations

3.1.1 The repository system used by ENRESA

ENRESA reference concept in clay is based on the disposal of spent fuel in carbon steel canisters in long horizontal disposal drifts excavated by means of tunnel boring machines. Canisters are surrounded by bentonite, a swelling clay. Galleries have a concrete liner made of wedged blocks. Access is accomplished by means of "main drifts" which run perpendicular to the disposal drifts. The main drifts meet at a central area, which includes the required underground infrastructure. Communication between the surface and the central underground area is accomplished by means of three access shafts and a ramp (ENRESA 2003).

The disposal drifts are located in a clay layer at a depth of 260 m. There are 100 m of clay above, 280 m of clay below the repository level, and 110 m of marls above the clay formation. On top of the marls there is a 50 m thick aquifer.

The carbon steel canister measures 4.54 m in length and 0.90 m in diameter, and contains four PWR or twelve BWR fuel elements in a subcritical configuration. The thickness of the wall of the canister is 0.10 m and is capable of withstanding the pressures to which it will be subjected under disposal conditions and of ensuring a minimum period of containment of one thousand years (although recent data point to a duration of several tens of thousands years). After being unloaded from the reactor, the fuel elements are temporarily stored during 50 years to allow the thermal power to decay to a level at which they may be disposed of, with a total thermal power of 1,200 W per canister. A total of 3,600 canisters will be required for the total waste inventory of spent fuel estimated for the Spanish nuclear programme.

Canisters are disposed in cylindrical disposal cells, built with blocks of pre-compacted bentonite. The disposal drifts are 500m long and have a concrete liner made of wedged blocks of 30 cm of thickness, resulting on an internal diameter of 2.4 m. Separations of 2.5 m between canisters and 50 m between disposal drifts have been established, in order not to exceed a temperature of 100ºC in the bentonite.

Once a disposal drift is filled with canisters and bentonite blocks, it is sealed with a 6 m long seal made of bentonite blocks and closed with a concrete plug at its entry. After completion of all the disposal drifts, main drifts, ramp, shafts and other remaining rock cavities will be backfilled with the clay extracted during the repository excavation. Seals of bentonite and concrete will be emplaced at regular intervals in the main drifts, ramp and shafts.

Due to the high suction pressures in the bentonite it will take a few hundred years at most to fully saturate the bentonite. When water contacts the canister a process of anaerobic corrosion starts. The minimum container lifetime due to generalized corrosion is around 10,000 years and the 3,600 canisters are assumed to fail sequentially at a constant rate during a time period equal to 20% the minimum container lifetime. After container failure, and since no credit is give to the cladding as a barrier, water reaches the irradiated UO_2 and radionuclide release starts.

The radionuclides and stable isotopes released from the waste pass to the water inside the canister, where they dissolve or precipitate depending on their solubility limits. Dissolved radionuclides are transported through the bentonite, the clay formation and the marls and finally reach the upper or lower aquifers.

The following processes are included in the transport model: Precipitation/redissolution, diffusion through bentonite, clay and marls, sorption in bentonite, clay and marls, advection through the clay formation and radioactive decay/ingrowth.

The data and models used in the transport calculations are described in the individual report of ENRESA (http://www.ip-pamina.eu). The model represents one single canister and the drift and host formation that corresponds to a single canister (50 m x 7.04 m surface). Nine different compartments are defined: Waste, Precipitate, Bentonite, 20 m of clay close to the repository (10 m above and 10 m below), Clay upper (100 m of clay above the repository), Clay lower (90 m of clay below the repository), Clay bottom (180 m of clay below Clay lower), 110 m of Marls above the clay, and the upper aquifer above the marls.

In the calculations performed the critical group uses water from a well drilled in the upper aquifer to produce all their aliments.

The radionuclide concentration in the well water is calculated dividing the radionuclide flux leaving the geosphere by the groundwater flow in the aquifer above the repository (180,000 m^3/a). The resulting concentration is multiplied by the dose conversion factors to obtain the doses. Radiotoxicity concentrations in water, radiotoxicity in compartments or radiotoxicity fluxes between compartments are calculated using the ingestion dose coefficients on the basis of ingestion dose coefficients for an adult (ICRP 1996).

3.1.2 The repository system used by NRG

In the TRUCK-II design (Barnichon et al. 2000), the reference repository is located at a depth of 500 m in a 100 m thick clay layer of the Boom Clay formation and includes shafts, galleries, disposal cells and other open volumes (e.g. workplaces). In the original TRUCK-II layout, galleries are surrounded by small horizontal disposal cells each enclosing a single

COGEMA container of vitrified HLW. For this study, the geometry of the disposal cell is modified to store one “Supercontainer” following a recent Belgian concept (De Bock et al. 2004), where a Supercontainer consists of two COGEMA containers surrounded by a large concrete buffer (1.9 m diameter). This results in a disposal cell of 6.5 m depth (including a sealing plug of 3 m) and with a diameter of 2 m.

When all containers are stored, the access galleries and other open volumes will be backfilled and the shafts will be sealed.

In the present study, the so-called “abandonment” scenario is analysed. The geological disposal of radioactive waste is a long-term operation that will be realized in subsequent stages, leading to extended time periods in which the facility is open and accessible. Political, societal or technical problems may lead to an abandonment of the facility. In case the shafts and galleries have not or only inaccurately been sealed, the repository may get flooded. As a consequence, preferential water-filled flow paths are formed that may lead to a significantly faster transport of radionuclides to the geosphere.

To represent a situation of abandonment, the disposal cell is connected to an open gallery, flooded and without backfilling material. The gallery is assumed to be directly connected to overlying aquifers. The model is established in two-dimensional cylindrical coordinates. Only one single disposal cell has been modelled, containing the inventory of one Supercontainer. The barriers formed by the stainless steel container, the steel overpack and the outer liner are not taken into account. Only diffusive transport of radionuclides is modelled, using sorption parameters from the BENIPA project (Sillen 2002). Precipitation of radionuclides is not taken into account.

The grid size of the computational domain of the gallery in x direction is taken equidistant with lengths of 1 m (0.35 m in the disposal cell, 0.1 m in the plug). The cell size in the gallery in r direction is 0.125 m. To reduce the calculation efforts, the cell sizes in the clay layer in r direction increases from 0.0125 m to 1 m at the top of the domain. The model discretization is tested for numerical artefacts.

3.1.3 The repository system used by SCK·CEN

The waste considered is the uranium oxide spent fuel that is expected to arise from the present Belgian nuclear programme, which consists of seven nuclear power plants of the PWR type with a total energy production of 5.5 GW and an expected reactor lifetime of 40 years. The equivalent amount of spent fuel is estimated to be 4,235 tHM (630 tHM has been reprocessed) with a burn up of 50 GWd/tHM.

The envisaged repository consists of a series of rectilinear galleries, in which the waste will be disposed, situated around the middle of the Boom Clay layer (at approximately 230 m depth below surface in case of the Mol site). Access to the disposal galleries is provided via the centrally located access gallery. The access gallery is connected to the surface via at least two shafts. Because the Boom Clay is a plastic clay, a concrete lining is required to avoid convergence of the gallery walls.

A key component of the engineered barrier system for spent fuel is the so called "supercontainer". The concept is aimed at establishing and maintaining a chemical environment around the overpack that should guarantee the required lifetime of the carbon steel overpack. In one supercontainer, four uranium oxide spent fuel assemblies are placed within a 30 mm thick overpack. The overpack will be placed into a prefabricated cementitious cylindrical buffer, possibly enclosed by a 6 mm thick stainless steel envelope. The buffer concrete will be based on ordinary Portland cement, which will provide the highly alkaline environment to keep the overpack passivated. The cylindrical cavity between the overpack and the buffer will be filled with a concrete filler. The top of the buffer is closed by pouring concrete which forms the sealing plug. These integrated waste packages are manufactured in surface buildings.

The distances between neighbouring supercontainers and between neighbouring disposal galleries are determined by thermal criteria and compatibility of waste types. The spent fuel supercontainers will be placed end-to-end with a pitch of 120 m between neighbouring disposal drifts. To avoid the adverse effects of the inevitable cave-in of the gallery walls over longer periods of time, void spaces are backfilled with a cementiteous material as much as possible.

The repository evolution scenario that is considered in the analysis is the expected evolution scenario. After perforation of the overpack, which is assumed to occur after 5,000 years, groundwater comes in contact with the waste form. The radionuclides that are not embedded in a stable waste matrix, i.e. the instantaneously released fraction, which corresponds to the gap inventory in the case of spent fuel disposal, can immediately dissolve in the pore water. In contact with groundwater, the waste matrices will start to corrode and consequently radionuclides are released from the waste matrix in the pore water. The fuel claddings and the structural metallic parts are assumed to corrode at a constant rate during 10,000 years. For the corrosion of the UO_2 matrix it is assumed that the corrosion rate is constant during a 1 million years period. When the concentration of a given radionuclide in the groundwater exceeds its solubility, that radionuclide will precipitate.

The main barrier of the repository system is the clay barrier provided by the host formation. Because of the low hydraulic conductivity of clay and the low hydraulic gradient over the host formation, the radionuclide transport through the clay layer is essentially due to molecular diffusion. The slow diffusive transport is for many radionuclides further retarded by sorption upon clay minerals. For the transport calculations the thickness of the clay barrier is taken equal to 40 m. The radionuclides that have migrated through the host clay formation reach the surrounding aquifers. As the overlying Neogene aquifer is used at present as a drinking water resource, the pumping of drinking water from a well located in this aquifer is considered. The well is assumed to be located just downstream from the border of the repository. At this location occur the highest concentrations in the aquifer water. The concentrations in the water pumped from a well located in the Neogene aquifer is calculated by using a dilution factor of 165,000 m^3/a. The Neogene aquifer is locally drained by the Kleine Nete river. The dose to a member of the critical group is calculated by using the results of the biosphere modelling. The considered critical group is a small self-sustaining community that uses water from a water well sunk in the Neogene aquifer, or from the Kleine Nete river, for drinking, watering of cattle and irrigation of crop fields and pastures. The considered exposure pathways are ingestion of water and food produced on the site, inhalation and external exposure.

3.2 Repositories in rock salt

3.2.1 The repository system used by GRS

The reference concept of the GRS contribution is based on a conceptual repository design proposed in Buhmann et al. (2008a). In this report the most important features are delineated; a description in more detail is provided by Buhmann et al. (2008b), Wolf et al. (2008) and in the GRS contribution for WP 3.4 (http://www.ip-pamina.eu).

The host rock formation is a salt dome with a sedimentary coverage of about 300 m. The repository is located in a depth of 870 m below surface (disposal level) in a homogeneous rock salt layer within the salt dome. The repository model consists of an access shaft, a central field and two access drifts, which connect the central field with a horizontal network of transfer drifts. From the inner transfer drifts boreholes are drilled to a depth of 300 m, 290 m are intended for emplacing the waste canister and 10 m for a plug. In the applied concept crushed salt with an initial porosity of 0.3 is applied for backfilling and for the borehole plug.

The strategy of the safety concept is the isolation of the emplaced waste by the tight and long-term stable rock salt formation. The waste canisters are not assumed to represent a long-term barrier. The function of the engineered barriers is to reseal the disturbed salt formation after the construction of the repository. The main engineered barriers are the shaft seal and the drift seals. The drift seals are located between the central field and the access drifts.

The repository is able to emplace the total waste volumes of HLW, which are expected to accumulate in Germany until 2080 (Bollingerfehr et al. 2008). Considered types of waste are spent fuel rods (SF), vitrified waste (HLW) and compacted constituents of spent fuel elements (ILW). The corresponding containers are thin-walled canisters (type BSK 3) for SF with a length of 4.98 m and a radius of 0.22 m, HLW canisters (type CSD-V, length = 1.34 m, radius = 0.22 m) and ILW canisters (type CSD-C, length = 1.35 m, radius = 0.22 m). Altogether 6,960 SF canisters, 3,225 HLW canisters and 7,455 ILW canisters are emplaced in the conceptual repository. The inventories of the different waste container types are based on data given in Buhmann et al. (2008b).

In the expected evolution no brine reaches the emplaced waste. For the calculation of the indicators it is assumed that the initial permeability of the shaft seal is only valid for a period of 75 years. After this period the permeability increases from 10^{-18} m^2 to 10^{-14} m^2. The permeability of the drift seals is set to a constant value of 10^{-15} m^2. This pessimistic parameter combination allows a brine inflow to the emplaced waste.

The failure of the canisters starts as soon as brine flows into a borehole with emplaced waste. For all canister types a uniformly distributed canister lifetime is assumed in the range between 0 and 10 years. The release from the waste matrix starts immediately after failure of the canister. Different mobilisation rates are used for the three types of waste. The corresponding mobilisation approaches for SF, HLW and ILW are discussed in Buhmann et al. (2008b).

The mobilised radionuclides are dissolved in the available water volume of the borehole. The radionuclides may precipitate if they reach their solubility limits within this water volume. Conservative solubility limits are used for the mobilisation process. Neither temporal change nor spatial differences in chemical conditions are taken into account. Sorption is not considered for the radionuclide transport in the repository.

For the radionuclide transport outside the repository advection, dispersion and dilution, sorption and radioactive decay are taken into account. The parameter values for the aquifer are based on investigations of overlying rocks of salt domes in Northern Germany. The geological formation along the migration pathway is modelled as a homogeneous medium with a porosity of 0.2 and an average width of 820 m and a thickness of 45 m. With a pore velocity of about 6.5 m/a, the resulting natural groundwater flow is 48,000 m^3/a.

The application of the biosphere model is in compliance with German regulations. It is assumed that the contaminated groundwater is used for irrigation, animal watering and drinking water. For the conversion of the radionuclide activity in the upper aquifer to the effective dose rate the following exposition pathways are included: consumption of contaminated drinking water, consumption of fish from contaminated ponds, consumption of plants irrigated with contaminated water, consumption of milk and meat from cattle watered and fed with contaminated fodder and exposure due to habitation on the contaminated land.

3.2.2 The repository system used by NRG

The hypothetical repository design in salt rock for HLW and MLW according to the EVEREST concept (Becker et al. 1997) is located at a depth of 840 m in a salt dome. The hypothetical site has a horizontal extension of 14 x 4 km^2. The central field consists of shafts, underground working places, warehouses, and galleries which lead to the boreholes and the chambers. Upon the termination of the disposal operations, the central field will be backfilled.

Vitrified HLW, packed in COGEMA containers, is stored in 194 vertical boreholes according to the Torad-B concept (Poley 2000). The boreholes are 500 m deep and are supported by vertical liners to enable a retrieval of the canisters as long as the facility is in operation. Every borehole can store about 300 COGEMA containers and is sealed by a 10 m thick plug of pre-compacted salt grit. After the placement of all canisters, all connecting galleries are backfilled with crushed rock salt and the repository is getting sealed by a dam. Due to creep behaviour of rock salt, the rock salt plug and the backfill will be impermeable within several thousand years.

In the present study, the so-called “brine-intrusion” scenario is analysed. In a salt dome, brine filled pockets exists, containing brine sealed in since million of years due to the impermeability of the surroundings rock salt. When such a brine pocket is located close to the repository, brine may enter the repository. For the analysed scenario, it is assumed that after proper closure of the repository, all open pore spaces that remains in the backfilled central field is getting filled by brine.

For the analysis of the scenario, the PA computer code EMOS is used. Several modules of the original EMOS code have been modified by NRG, mostly with regard to the modelling of the compaction behaviour of salt grit and the porosity-permeability relation (Schröder 2008). In the analysis performed, only one borehole is modelled, connected to the geosphere by the “Central field”, a volume representing all galleries, stores, workplaces or other open volumes in the repository. A container-model describes the dissolution of the vitrified waste in the COGEMA-containers.

The modelling of the far-field is done by the geosphere-model MASCOT consisting of a set of transfer-functions that calculates the transport and mixing of brine outside the repository. The biosphere-model EXPOS consist of a set of dose-conversion factors, derived for the specific environment of the repository, for the estimation of the potential exposure to radionuclides in the biosphere via different exposure routes.

3.3 Repositories in granite

All calculations were performed for the current Czech concept for a deep geological repository. This concept is similar to the KBS-3 concept of disposal of spent fuel assemblies in granite in a vertical position (EGP 1999, Lindgren et al. 1999). The repository layout in granite host rock consists in spent fuel waste packages surrounded by bentonite bricks and located in the tunnels at least 500 m under the surface. Instead of copper based canisters proposed in KBS-3 concept, carbon steel canisters were proposed for assemblies from VVER reactors. This concept is now, however under review in a new project initiated recently by RAWRA. Also the possibility of using copper canisters will be considered.

Modelling calculations have been performed for spent fuel assemblies from NPP Dukovany. It was supposed that the spent fuel assemblies (SFAs) would be disposed in 1,440 canisters each containing seven spent fuel assemblies Average inventory in SFAs corresponds to burn up of 35,000 MWd/tU. Canisters with SFAs are surrounded by bentonite bricks with thickness of 30 cm. No credit was taken for transport constraints of failed canisters in calculations.

The following assumptions were accepted for calculations:

- For 1,300 years all the carbon steel canisters are intact and no radionuclides can escape from the waste packages. The only processes modelled in this time are radioactive decay and ingrowth of radionuclides.
- After this time, the canisters will start to fail due to the corrosion of canisters and the effect of surrounding pressure. The failure of canister will follow a Weibull-distribution.
- Water contacts the waste, which begins to dissolve, and radionuclides start to escape. In the spent fuel, three locations of nuclides are considered: gap, waste matrix and structural materials. Nuclides in the gaps are assumed to be released to the canister zone immediately after contact with water. The release from the matrix is modelled by a simple leaching process, the amount released in unit time being proportional to the mass remaining in matrix.
- It is assumed that in spent fuel assemblies 1 – 10% of mobile radionuclides (I-129, Cl-36, Se-79, C-14, etc.) are located in so-called gap fraction immediately free for transport after contact with water.
- Radionuclides can precipitate due to exceeding their solubility limits and diffuse into bentonite.
- In the bentonite, radial diffusive transport driven by the concentration gradient between the canisters zone and the host rock is modelled. Radionuclides also sorb onto buffer material. The outer edge of the backfill will have a mixing cell condition given by the product of external water flow around buffer and concentration at the exit of buffer.

The concept of stream tubes is used for representation of geosphere. The calculations have been performed in compartment code GoldSim version 9.6.

4. Safety indicators

One important aspect of WP 3.4 is the development and testing of safety indicators complementary to dose and risk. These additional indicators are required to strengthen the robustness of the safety case by using multiple lines of evidence and to compensate the shortcomings of single indicators.

In WP 3.4 three additional indicators were identified that could contribute to a higher confidence in the safety statements given by a post-closure safety analysis:

- the radiotoxicity concentration in biosphere water,
- the radiotoxicity flux from the geosphere and
- the contribution to the power density in ground water.

The complementary safety indicators “radiotoxicity concentration in biosphere water” and “radiotoxicity flux from the geosphere” were already discussed and applied for a granite formation in the SPIN project. In contrast to these indicators the indicator “contribution to the power density in groundwater” is a new proposal suggested by Baltes et al. (2007).

In section 4.5 safety indicators based on the concept of risk are presented.

4.1 Effective dose rate

The individual dose rate represents the annual effective dose to an average member of the group of the most exposed individuals. It takes into account dilution and accumulation in the biosphere, different exposure pathways as well as living and nutrition habits. It is calculated using

$$\sum_{\text{all nuclides } n} c_n B_n$$

with the activity concentration c_n [Bq/m^3] of radionuclide n in the biosphere water, which is used by man for drinking, feeding livestock or irrigation and which receives the releases of radionuclides from the geosphere. The biosphere dose conversion factor B_n, is the annual dose to the receptor caused by a unit concentration of radionuclide n in the biosphere water. It is measured in [(Sv/a)/(Bq/m^3)]. The effective dose takes into account several exposition pathways and is widely used. The safety statement of the indicator has a clear relevance for human health. The biosphere water is assumed to be taken from a well in a near-surface aquifer or a river.

The effective dose rate is internationally accepted as the main indicator for assessing the safety of a repository system. In many countries the regulatory authorities have established regulatory limits for this indicator. In general these limits represent a small fraction of the natural background radiation doses, which is in Europe in the range of 2 to 3 mSv/a.

The applied reference value of all participating groups for the effective dose rate is **0.1 - 0.3 mSv/a**.

4.2 Radiotoxicity concentration in biosphere water

This indicator represents the radiotoxicity of the radionuclides in 1 m^3 of biosphere water. It is calculated as

$$\sum_{\text{all nuclides } n} c_n D_n$$

with the ingestion dose coefficient D_n, which represents the dose caused by ingestion of radionuclide n (Sv per ingested Bq). Ingestion dose coefficients of ICRP (1996) are used. The effects of daughters produced in vivo are accounted for in the ingestion dose coefficients.

For the computation of the radiotoxicity concentration in the biosphere water no exposure pathways need to be defined. The radiotoxicity concentration is thus independent of the biosphere, and can therefore be regarded as a more robust indicator for longer time frames. In comparison to the individual dose rate the safety statement of this indicator is restricted in a way that it assesses only the integrity of the drinking water from the contemplated aquifer with respect to human health.

Reference values can be obtained for instance by computing the natural radiotoxicity of groundwater, based on the actual concentrations of naturally-occurring radionuclides (ENRESA, SCK·CEN, NRG, NRI). GRS used the radionuclide concentrations in drinking water in Germany for the derivation of the reference value. These values are used since it can be assumed that observed concentrations in drinking water are harmless for human health. The reference value derived by GRS is one order of magnitude lower than the other values.

4.3 Radiotoxicity flux from the geosphere

While the radiotoxicity concentration in the biosphere water is an indicator that attempts to avoid the uncertainty associated with future biospheres, it is still affected by the uncertainty in the dilution in the considered water body. To avoid the latter uncertainty, one can use the radiotoxicity flux[1] from the geosphere, which can be defined without using dilution in aquifers or rivers. However, whereas the calculated indicator is less dependent on uncertain data, the uncertainty is transferred to the estimation of relevant reference values that can be used for comparison. It is also very important to clearly define the extent of the geosphere, in order to avoid confusion.

The radiotoxicity flux from the geosphere is calculated as

$$\sum_{\text{all nuclides } n} s_n D_n$$

with the activity flux s_n [Bq/a] of radionuclide *n* from the geosphere to a water body, which represents the interface to the biosphere. The radiotoxicity flux eliminates the uncertainty from the dilution in this water body, but it has only a weak relation to human health. It is preferably applicable to long time frames. The safety statement of this indicator is that there is no significant influence of the repository on the radiotoxicity of the water body (for instance the water in an upper aquifer or a river).

The derivation of the reference value for the safety indicator radiotoxicity flux from the geosphere is more problematic than for the effective dose and the radiotoxicity concentration in biosphere water. A reference value for this indicator is usually site-specific since it is based on the natural groundwater flow, which cannot be determined on very large scales. But even on local scales the determination of a natural groundwater flow in the vicinity of a repository is difficult and implies a lot of uncertainties. Due to these reasons the range of applied reference value is much larger than for the other reference values.

4.4 Power density in ground water

The indicator “power density in groundwater” is a physical parameter independent of any specific biological species. It is composed of the contribution of all radionuclides and can be seen as a criterion for the impact of hazardous radionuclides on biota in general. But since the radiological consequences can not be assessed by this indicator, it has only a limited relevance for safety compared to the other presented safety indicators. Nevertheless, the information embedded in this indicator can be very useful as an additional safety argument in a safety case.

To obtain numerical values for this parameter the decay energy, i.e. the total disintegration energy of an individual decay process (unit [MeV]), must be determined for every considered radionuclide. The sources for the applied decay energy are Firestone (1999) and Weast (1986).

[1] In environmental sciences a flux is in general a magnitude that flows through a unit area per unit time. Other disciplines use the term flux for a magnitude that represents already an integral of such an amount over a specific surface. In this sense term radiotoxicity flux has been used in the radwaste community for a long time and is used in this report.

The calculation of the power density is carried out with a simple weighting scheme by multiplying the activity concentration of every radionuclide [Bq/m^3] with its decay energy. This operation yields a power density p (power per volume, [MeV/(s·m^3)])

$$p = \sum_{\text{all nuclides } n} c_n E_n$$

with the activity concentration c_n [Bq/m^3] of radionuclide n in groundwater and the corresponding decay energies E_n.

Originally, it was proposed to use the power density in the pore water and in the soil matrix in the deeper aquifer system (Baltes et al. 2007). But so far no reference value has been derived for such a system. Measuring the necessary data for this system is quite difficult and out of the scope of the PAMINA project. Therefore a different reference was used to compare naturally existing power densities to power densities released from a repository system. As an example, GRS calculated the natural background power density in the upper aquifer of the Gorleben salt dome (chapter 5.4). This value was also used by ENRESA and NRG for their calculations.

4.5 Safety indicators based on risk

The uncertainties associated with the evolution of the disposal system must be appropriately considered. One important tool of managing and communicating these uncertainties is the calculation of the risk emanating from a repository. Risk in general means the probability that a specified undesirable event will occur in a specified period or as a result of a specified situation (HSE 1988).

For a quantitative analysis it has to be more clearly defined, what risk means. The general statistical-mathematical formulation defines risk (*R*) as a combination (the product) of the probability of the occurrence of an adverse event for a defined time frame and a measure of the consequences of this event.

The dose risk R_D (in general risk is related to the effective dose rate) resulting from a number of possible scenarios with the probabilities p_i can be calculated as

$$R_D = \sum_{\text{all scenarios } i} p_i C_i \ ,$$

where C_i is consequence (here the dose rate) resulting from scenario *i*. Consequently, the dose risk is measured in Sv/yr. For a complete description of the future of a repository, the scenario probabilities must add up to 1.

Despite this simple mathematical description of risk, the actual risk originating from a repository is hard to determine. The main problem is to determine the number of scenarios considered to be relevant in the assessment, the conditions of these scenarios and their probabilities.

In order to compare the calculated dose risk with other accepted risks it must be transformed to a more illustrative value that allows assessing the individual risk. For this purpose an indicator based on risk is introduced, which compares the probability of an event and its consequences (e.g. the risk of a certain effective dose R_D) with a risk limit accepted by regulators and/or the society. To compare the consequences with such a measure a quantitative evaluation of the incidence of an adverse effect that is expected in a population as a result of an exposure to these consequences is necessary. For a simple linear dose-response-function without a threshold value this can be done by defining a risk coefficient r

$$R = rR_D$$

Assuming this linear relationship, risk-per-dose coefficients r between about 0.04 Sv^{-1} and 0.07 Sv^{-1} have been found, depending on the group of people considered and whether only fatal or all kinds of cancer are considered. In ICRP (2008) a value of r = 0.05 Sv^{-1} to 0.07 Sv^{-1} is recommended.

For a comparison with risks from other natural or technical sources it is very important to clearly define the risk measures that are used in the analysis. A lot of comparisons are possible, but is often difficult to get reliable numbers. In WP 3.4 two well determined technical risks are applied for a comparison with the risk emanating from the repository system:

- The risk of having a fatal accident in road traffic in Germany. The derived risk is about **$6 \cdot 10^{-5}$ per year** in the reference year 2007.
- The risk of dying in a plane crash. The derived risk is about **$1 \cdot 10^{-7}$ per year**.

The derivation of the reference values is described in the individual report of GRS. Both risks deal with means of transport humans have to use in everyday life. These risks are necessary technical risks and are generally accepted by the majority. Furthermore, they are relatively easy to determine.

Besides these two reference value a third comparison value for the risk emanating from a repository system is applied here. It is called the acceptable risk. That is the level of loss a society considers acceptable given existing social, economic, political, cultural, technical and environmental conditions. In environmental and especially in nuclear sciences there is the general agreement, that a risk of **$1 \cdot 10^{-6}$ per year** of suffering a serious health effect is an appropriate level as a regulatory constraint or target, e.g. HSE (1988) or NEA (2005).

5. Determination of reference values

To evaluate the calculated results for safety indicators, the end points of the safety assessment must be compared with established reference values that indicate an acceptable level of safety. If a safety indicator is below the corresponding reference value, it can be stated that the repository is “safe” with regard to a particular safety aspect.

It is widely accepted, that ”reference values from natural systems provide sensible yardsticks for disposal facility-derived concentration and fluxes” (IAEA 2003). For this reason reference values for the safety indicators are usually derived from natural conditions that are considered by the corresponding regulatory bodies as “safe” for the population.

Four participants worked on the derivation of possible reference values for safety indicators in WP 3.4. The main results obtained by Amphos21, SCK·CEN, NRI and GRS on this topic are presented in the next sections. Section 5.5 gives a summary of all reference values applied in WP 3.4.

5.1 Derivation of reference values for the radiotoxicity concentration in groundwater and flux from geosphere (Amphos21)

Amphos21 has performed an extensive bibliographic research to identify natural radionuclide concentrations in the aquifer groundwater of different types of geological formations all over the world: crystalline rocks, karst, limestone, sand and sandstone. The maximum and minimum natural concentrations of each radionuclide are presented in Table 5.1. More details can be found in the individual report of Amphos21.

The radiotoxicity concentrations in the groundwater are the product of the natural activity concentration times the dose conversion factor by ingestion for an adult provided by ICRP (1996). The resulting radiotoxicity of the natural waters due to each radionuclide is also presented in Table 5.1.

The radiotoxicities of all the radionuclides are added to obtain the total radiotoxicity concentration in natural groundwater (Table 5.2), and the range of values obtained is from 10^{-8} to 10^{-3} Sv/m^3. On the basis of these data a reference value of 10^{-5} Sv/m^3 is recommended to be used in calculations. This value is very similar to the $2 \cdot 10^{-5}$ Sv/m^3 obtained in SPIN.

Natural radiotoxicity fluxes are site-specific and the methodology to derive reference values for radiotoxicity fluxes is not straightforward. For a particular site, a reference value can be obtained using the groundwater flows obtained with the hydrogeological modelling and multiplying by the reference value for the radiotoxicity concentration in natural waters. Following this approach, if the natural groundwater flow is 10^5 m^3/a, the reference value for the radiotoxicity flux from the geosphere would take a value of 1 Sv/a.

Table 5.1 Radionuclide concentration and radiotoxicity in natural groundwater

Isotope	Aquifer type	Concentration (Bq/l)		Radiotoxicity (Sv/m^3)	
		Minimum	Maximum	Minimum	Maximum
^{40}K	Crystalline rocks	5.62E-03	7.86E-02	3.56E-08	4.98E-07
	Karst	9.27E-03	5.34E-02	5.87E-08	3.38E-07
	Limestone	1.12E-03	5.90E-02	7.10E-09	3.74E-07
	Sand	2.41E-02	1.45E+00	1.53E-07	9.19E-06
	Sandstone	1.38E-02	6.18E-01	8.74E-08	3.92E-06
^{223}Ra*	Crystalline rocks	5.76E-05	1.88E-01	5.89E-09	1.92E-05
	Karst			0.00E+00	0.00E+00
	Limestone	6.72E-09	4.03E-04	6.87E-13	4.12E-08
	Sand	1.73E-06	1.10E-03	1.77E-10	1.12E-07
	Sandstone		8.79E-03	0.00E+00	8.98E-07
^{224}Ra*	Crystalline rocks	1.58E-08	1.60E-03	1.05E-12	1.06E-07
	Karst			0.00E+00	0.00E+00
	Limestone			0.00E+00	0.00E+00
	Sand	1.00E-04	4.00E-04	6.64E-09	2.66E-08
	Sandstone		1.60E-04	0.00E+00	1.06E-08
^{226}Ra	Crystalline rocks	4.00E-06	5.29E-03	1.14E-09	1.51E-06
	Karst			0.00E+00	0.00E+00
	Limestone	3.30E-06	1.30E-05	9.44E-10	3.72E-09
	Sand			0.00E+00	0.00E+00
	Sandstone	2.20E-06	2.95E-05	6.30E-10	8.44E-09
^{228}Ra*	Crystalline rocks	1.58E-08	1.60E-03	1.11E-11	1.13E-06
	Karst			0.00E+00	0.00E+00
	Limestone			0.00E+00	0.00E+00
	Sand	1.00E-04	4.00E-04	7.05E-08	2.82E-07
	Sandstone		1.60E-04	0.00E+00	1.13E-07
^{222}Rn	Crystalline rocks	1.75E-07	7.11E-05		
	Karst				
	Limestone	2.46E-07			
	Sand				
	Sandstone	5.79E-07			
^{87}Rb	Crystalline rocks	1.78E-04	4.01E-03	2.73E-10	6.15E-09
	Karst			0.00E+00	0.00E+00
	Limestone	8.91E-05	1.07E-03	1.37E-10	1.64E-09
	Sand	6.86E-04	4.29E-02	1.05E-09	6.58E-08
	Sandstone	1.46E-03	3.94E-02	2.24E-09	6.04E-08
^{228}Th*	Crystalline rocks	1.58E-08	1.60E-03	1.16E-12	1.18E-07
	Karst			0.00E+00	0.00E+00
	Limestone			0.00E+00	0.00E+00
	Sand	1.00E-04	4.00E-04	7.36E-09	2.94E-08
	Sandstone		1.60E-04	0.00E+00	1.18E-08

Isotope	Aquifer type	Concentration (Bq/l)		Radiotoxicity (Sv/m^3)	
		Minimum	Maximum	Minimum	Maximum
^{230}Th*	Crystalline rocks	1.19E-03	3.89E+00	2.55E-07	8.35E-04
	Karst			0.00E+00	0.00E+00
	Limestone	1.39E-07	8.34E-03	2.98E-11	1.79E-06
	Sand	3.57E-05	2.28E-02	7.66E-09	4.89E-06
	Sandstone		1.82E-01	0.00E+00	3.91E-05
^{232}Th	Crystalline rocks	1.58E-08	1.60E-03	3.71E-12	3.76E-07
	Karst			0.00E+00	0.00E+00
	Limestone			0.00E+00	0.00E+00
	Sand	1.00E-04	4.00E-04	2.35E-08	9.40E-08
	Sandstone		1.60E-04	0.00E+00	3.76E-08
^{234}Th*	Crystalline rocks	1.19E-03	3.89E+00	4.14E-09	1.35E-05
	Karst			0.00E+00	0.00E+00
	Limestone	1.39E-07	8.34E-03	4.83E-13	2.90E-08
	Sand	3.57E-05	2.28E-02	1.24E-10	7.92E-08
	Sandstone		1.82E-01	0.00E+00	6.32E-07
^{234}U*	Crystalline rocks	1.19E-03	3.89E+00	5.96E-08	1.95E-04
	Karst			0.00E+00	0.00E+00
	Limestone	1.39E-07	8.34E-03	6.96E-12	4.18E-07
	Sand	3.57E-05	2.28E-02	1.79E-09	1.14E-06
	Sandstone		1.82E-01	0.00E+00	9.11E-06
^{235}U	Crystalline rocks	5.76E-05	1.88E-01	2.77E-09	9.03E-06
	Karst			0.00E+00	0.00E+00
	Limestone	6.72E-09	4.03E-04	3.23E-13	1.94E-08
	Sand	1.73E-06	1.10E-03	8.31E-11	5.28E-08
	Sandstone		8.79E-03	0.00E+00	4.22E-07
^{238}U	Crystalline rocks	1.19E-03	3.89E+00	5.47E-08	1.79E-04
	Karst			0.00E+00	0.00E+00
	Limestone	1.39E-07	8.34E-03	6.39E-12	3.84E-07
	Sand	3.57E-05	2.28E-02	1.64E-09	1.05E-06
	Sandstone		1.82E-01	0.00E+00	8.37E-06

* Activity of the radionuclide equal to the parent activity (secular equilibrium)

Table 5.2 Radiotoxicity concentration in natural groundwater

Aquifer type	Radiotoxicity (Sv/m^3)	
	Minimum	Maximum
Crystalline rocks	4.19E-07	1.25E-03
Karst	5.87E-08	3.38E-07
Limestone	8.23E-09	3.06E-06
Sand	2.74E-07	1.70E-05
Sandstone	9.03E-08	6.27E-05

5.2 Reference value for the radiotoxicity flux from geosphere in Belgium (SCK·CEN)

It is not possible to derive a reference value for radionuclide fluxes from the available measurements of natural radionuclides in the candidate host formation (Boom clay), because the observed concentration profiles are very irregular. Instead, a reference value is derived from the data on the application of phosphate fertilisers in Flanders.

Table 5.3 gives the activity concentrations of U-238, U-234, Ra-226 and Th-230 in fertilisers, the activity fluxes for an application rate of 4760 kg/km², the dose factor and the resulting radiotoxicity fluxes.

Table 5.3 Activity concentrations in fertilisers, activity fluxes for an application rate of 4760 kg/km², the dose factor and the resulting radiotoxicity fluxes

	Activity concentration		Activity flux		Factor	Radiotoxicity flux	
	Minimum	Maximum	Minimum	Maximum		Minimum	Maximum
	(Bq/kg)	(Bq/kg)	(Bq/a/km²)	(Bq/a/km²)	(Sv/Bq)	(Sv/a/km²)	(Sv/a/km²)
U-238	300	3000	1428000	14280000	4.84E-08	6.91E-02	6.91E-01
U-234	300	3000	1428000	14280000	4.90E-08	7.00E-02	7.00E-01
Ra-226	200	1000	952000	952000	2.17E-06	2.07E+00	1.03E+01
Th-230	300	3000	1428000	14280000	2.10E-07	3.00E-01	3.00E+00
Total	1100	10000	5236000	47600000		2.50E+00	1.47E+01

The radiotoxicity flux that is spread every year on an agricultural field ranges between 2.5 and 15 Sv/km².

Assuming that a typical size of a geological repository that can accommodate the spent fuel arising from the present Belgian nuclear programme is 1 km², a possible reference value for the radiotoxicity flux might be 10 Sv/a. For comparison, the reference value used in SPIN project was 60 Sv/a.

5.3 Reference values for radiotoxicity concentration and flux from geosphere in the Czech Republic (NRI)

The work performed by NRI was concentrated on the compilation of activity concentrations of groundwater and surface water monitored during the second half of 20th century by various Czech organizations:

- Czech Hydro-Meteorological Institute (surface and ground water radioactivity monitoring) http://www.chmi.cz

- Geofond - Czech Geological Survey (data collection from geological boreholes and wells) http://www.geofond.cz,

- National Radiation Protection Institute (monitoring of drinking surface and ground water quality) http://www.suro.cz.

Parameters measured were mainly gross alpha activity, gross beta activity and the activity concentration of Uranium, K-40 and Ra-226.

Gross measurements are used as a method to screen samples for relative levels of radioactivity so that only total alpha or total beta activity is measured, regardless of the specific radionuclide source. They were corrected for ingrowth activity coming from decay of Ra-226 and K-40. A summarization of minimum and maximum values of compiled and corrected data for surface water and groundwater is given in Table 5.4.

Radiotoxicity concentration intervals (gross alpha and gross beta) were calculated by multiplication of the derived minimum and maximum activity concentrations and minimum and maximum conversion factors for ingestion (Table 5.5). The derivation of the minimum and maximum conversion factors for ingestion is explained in the individual report of NRI.

Table 5.4 Activity concentrations of surface water and groundwater in the Czech Republic

Measured variable	Symbol	Unit	Surface water (Bq/m^3)		Groundwater (Bq/m^3)	
			Minimum	Maximum	Minimum	Maximum
Gross alpha	A_{α}	Bq/m^3	68	239	10	14,000
Gross beta	A_{β}	Bq/m^3	202	405	80	5,300
Uranium	C_U	g/m^3	0	0.012	0.0001	0.6
Ra-226	A_{Ra226}	Bq/m^3	0	250	5	1,200
K-40	A_{K40}	Bq/m^3	10	301	11	2,160
Uranium activity	A_U	Bq/m^3	0	30.8	0.256	1,538
Corrected gross alpha activity	$A_{\alpha C}$	Bq/m^3	68	0	4.74	11,300
Corrected gross beta activity	$A_{\beta C}$	Bq/m^3	192	104	69	3,140

Table 5.5 Calculated radiotoxicity concentration in surface and ground water

Variable	Factor (Sv/Bq)		Surface water (Sv/m^3)		Groundwater (Sv/m^3)	
	Minimum	Maximum	Minimum	Maximum	Minimum	Maximum
Gross alpha	6.50E-08	1.20E-06	4.42E-06	0	3.08E-07	1.36E-02
Gross beta	6.70E-07	1.10E-06	1.29E-04	1.14E-04	4.62E-05	3.45E-03
Uranium	4.50E-08	4.90E-08	0	1.51E-06	1.15E-08	7.54E-05
Ra-226	2.80E-07		0	7.00E-05	1.40E-06	3.36E-04
K-40	6.20E-09		6.20E-08	1.87E-08	6.84E-08	1.34E-05
Total			1.33E-04	1.88E-04	4.80E-05	1.74E-02

The radiotoxicity concentrations in groundwater (Table 5.5) were used then for calculations of radiotoxicity flux from geosphere (Table 5.6). The groundwater run-off values were taken mainly from work of Krásný et al. (1982).

Table 5.6 Radiotoxicity flux from the geosphere

	Units	Minimum	Maximum
Groundwater radiotoxicity	Sv/m^3	4.80E-05	1.74E-02
Groundwater run-off	m^3/a/km^2	1.58E+04	3.16E+05
Radiotoxicity flux from geosphere	Sv/a/km^2	7.57E-01	5.50E+03

It can be concluded that the range of reference values for deep geological repositories in the Czech Republic is very broad. The radiotoxicity concentration for surface water is in the range from $1.33{\cdot}10^{-4}$ to $1.88{\cdot}10^{-4}$ Sv/m^3 and for groundwater in the range from $4.80{\cdot}10^{-5}$ Sv/m^3 to $1.74{\cdot}10^{-2}$ Sv/m^3. The flux of radiotoxicity from geosphere is in the range from 0.757 to 5,500 Sv/a/km^2.

The reference value for the radiotoxicity flux from the geosphere (8 Sv/a) was obtained by multiplication of minimal natural radiotoxicity from the geosphere (0.757 Sv/a/km^2) with the expected surface area of the repository (10 km^2).

5.4 Reference value for the power density in groundwater (GRS)

The background value for the power density in groundwater is determined for the upper aquifer for the area at Gorleben, since a lot of data from an extensive drilling, exploration and monitoring programme of the Gorleben salt dome are available.

Concentrations for uranium and thorium are available from different wells of the upper aquifer. The concentrations of all other radionuclides are calculated from the concentration of the mother nuclides of the three natural decay chains U-238, U-235 and Th-232. It is assumed that all radionuclides in a decay chain are in secular equilibrium, i.e. the total activity concentration of each radionuclide in the decay chain corresponds to that of the mother nuclide.

The measured concentrations represent the mobile fraction of the total activity concentration. The mobile fraction of each radionuclide is determined by its sorption properties. These are different for the different elements. The total concentration of all radionuclides in a single decay chain is calculated as the product of the concentration of the respective mother nuclide in the groundwater $c_{l,m}$ and the retardation factor of the mother nuclide $R_{f,m}$. The mobile concentration of radionuclide i is then derived from

$$c_{l,i} = \frac{c_{l,m} \cdot R_{f,m}}{R_{f,i}}$$

with the retardation factor

$$R_f = 1 + \frac{(1-n)}{n} \rho_s K_d$$

the porosity n, the rock density ρ_s, and the element specific distribution coefficient K_d. The mother nuclide is denoted by the index m and the respective radionuclide from the same decay chain with index i.

The concentration of each radionuclide in the groundwater is determined by the calculation of the ratio of the retardation factors of the mother nuclide and the respective radionuclide.

Table 5.7 Radiotoxicity flux from geosphere

Radionuclide	Half-life [a]	Natural abundance in element [wt. %]	Conversion factor [Bq/kg]	Average conc. [Bq/m^3]
U-238	4.468E+09	99.2742	1.245E+07	2.12E+00
U-234	7.038E+08	0.7204	8.000E+07	9.88E-02
Th-232	1.405E+10	100.00	4.065E+06	1.35E+00

The mean groundwater concentrations for uranium and thorium are derived from data of 14 and 19 samples, respectively, which are available for the near surface aquifer in the Gorleben area. The resulting mean concentration is 0.72 nmol/l ($1.7 \cdot 10^{-7}$ kg/m^3) for uranium and 1.43 nmol/l ($3.3 \cdot 10^{-7}$ kg/m^3) for thorium. The activity concentration of the three radionuclides in the groundwater is calculated with nuclide-specific activity-to-mass conversion factors taking into account the natural abundance of the uranium isotopes. The resulting average activity concentrations are listed in Table 5.7.

The R_f-values for each radionuclide are taken from Suter et al. (1988). For radon no sorption data were used in this approach. Since all radon isotopes are volatile the assumption that radon is in secular equilibrium with its mother nuclides is probably not correct. In spite of its importance for the natural radioactivity it is neglected here. The advantage of this procedure is the provided low reference value. A low reference value enhances the confidence in the safety statement given by the corresponding safety indicator. For the same reason other natural radionuclides are not considered either.

The total natural power density in groundwater is calculated as the sum of the power densities of all three decay chains. The total natural power density in groundwater is then 78.3 MeV/(s·m^3).

5.5 Summary of the applied reference values

Table 5.8 presents the reference values for the safety indicators used in the calculations done by the participants in WP 3.4. The reference values have different sources, some of them

- represent legal limits,
- are taken from the SPIN-project,
- are derived within WP 3.4 (chapter 5.1 to 5.4) or
- from other investigations.

Table 5.8 Reference values for the safety indicators used in the calculations

	ENRESA	SCK·CEN	NRG	NRI	GRS
Effective dose rate (mSv/a)	0.1-0.3 [a]	0.3 [a]	0.3 [e]	0.25 [a]	0.1 [d]
Radiotoxicity concentration in biosphere water (Sv/m^3)	$2 \cdot 10^{-5}$ [b]	$2 \cdot 10^{-5}$ [b]	$2 \cdot 10^{-5}$ [b]	$4 \cdot 10^{-6}$ [c]	$2 \cdot 10^{-6}$ [d]
Radiotoxicity flux from the geosphere (Sv/a)	60 [b]	10 [c]	60 [c]	8 [c]	0.1 [d]
Power density in biosphere water (MeV/s·m^3)	80 [c]	-	80 [c]	-	25 [c]

[a] Legal limits

[b] Reference values derived in SPIN

[c] Reference values derived in PAMINA

[d] Reference values derived in Wolf et al. (2008)

[e] IAEA recommendations (IAEA 06)

The corresponding sources are given in Table 5.8. The reference values for the indicator “radiotoxicity flux from the geosphere” has the largest range, since they are derived for local conditions. The reference value for the “power density in groundwater” is derived from the conditions in the upper aquifer in the Gorleben area. In contrast to the other participants GRS used a safety margin (about one third of natural background value) for their reference values.

6. Safety indicator results

In this chapter the safety indicator results calculated for the different studies are presented and assessed. For better comparability the results are predominantly presented as normalised values, divided by the respective reference values. This allows direct comparison of the safety statements about a repository system that can be derived from the different indicators. This is considered a powerful method of communicating arguments for – or against – the safety of a system.

Where the absolute values of the calculated indicators are presented, this is for illustrating how the individual radionuclides contribute to the indicator, or for other specific purposes.

It should be clearly stated that it is not intended, and has never been an objective of the work described here, to compare different repository systems or different host rock types with regard to safety. Such a comparison would be misleading for the following reasons:

- The modelled repository systems are in different stages of development. Whereas the Boom clay has been the subject of a comprehensive characterisation programme, for most other sites a lot of data, parameters and model concepts are based on generic figures and approaches.

- The radioactive waste inventory is of different amount and/or nature

- The reference values are partly different for the considered repository systems

- Different kind of scenarios are analysed in the different studies (i.e. normal evolution scenario, brine intrusion scenario, abandonment scenario)

- There is no complete scenario development and discussion of scenario probabilities for the modelled repository systems. The different probabilities of the scenarios for the calculations are not taken into account.

- The ongoing (and difficult) discussion how to compare quantitatively repository systems in different host rock types is out of the scope of this working package.

In general, the temporal evolution of the considered safety indicators is quite similar for each of the simulation. The reason is that they are all based on the radionuclide flux from the repository and that only a few radionuclides dominate the release. The dominating radionuclides in the different systems are:

- ENRESA (clay): I-129, Se-79, Cl-36, and Ra-226

- NRG (clay): Se-79, Sn-126, and Th-229

- SCK·CEN (clay): Se-79, I-129, Cl-36, Sn-126, Tc-99, and Th-229

- GRS (salt): I-129, Cs-135, Cl-36, and Ra-226
- NRG (salt): Se-79, Cs-135, Ra-226, and Th-229
- NRI (granite): I-129, C-14, Sn-126, Cs-135, and Ra-226

The concept of NRG considers only reprocessed waste, therefore I-129 is not relevant.

In most recent safety cases results of the calculations are presented up to 1 million years. However, in case of disposal in clay releases of actinides from the clay layer occur after 1 million years. Therefore, SCK·CEN and ENRESA give here results up to 10 and 100 million years respectively.

The following sections summarise the main results of the individual reports of WP 3.4. Further general conclusions are drawn in chapter 10.

6.1 The Spanish concept in clay (ENRESA)

For the Spanish concept ENRESA has tested the selected indicators both in probabilistic and deterministic calculations.

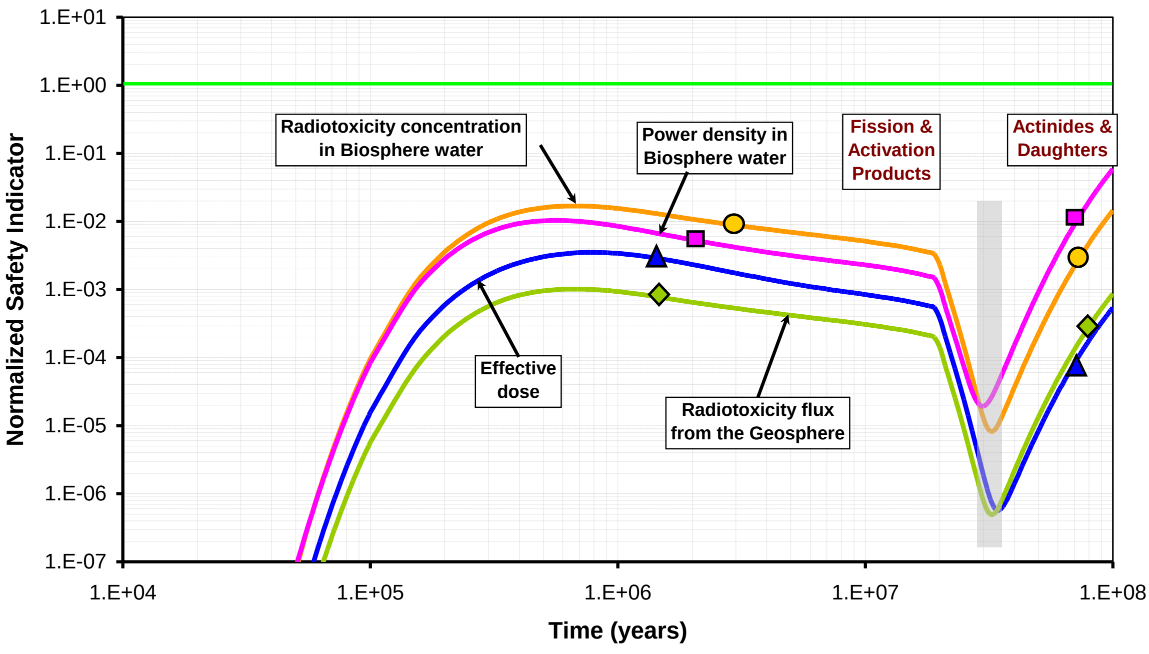

Figure 6.1 The normalised safety indicators calculated by ENRESA.

The normalised safety indicators in the deterministic calculation are shown in Figure 6.1. The four curves have similar shapes and are controlled by fission and activation products up to 30 million years and by actinides and daughters (4n+2 series) thereafter. For doses the reference value is a regulatory limit (0.1 mSv/a) while for the other safety indicators natural values are used (see chapter 5).

Figure 6.1 shows that the importance of the long-term effects (after 10 million years) due to actinides and daughters compared with the shorter term effects of the fission and activation products is higher in terms of radiotoxicity than in terms of doses, and is even greater in terms of power density. While peak dose is controlled by fission and activation products (mainly I-129), peak radiotoxicity concentrations and fluxes are roughly the same for both groups and the peak power density is controlled by actinides and daughters.

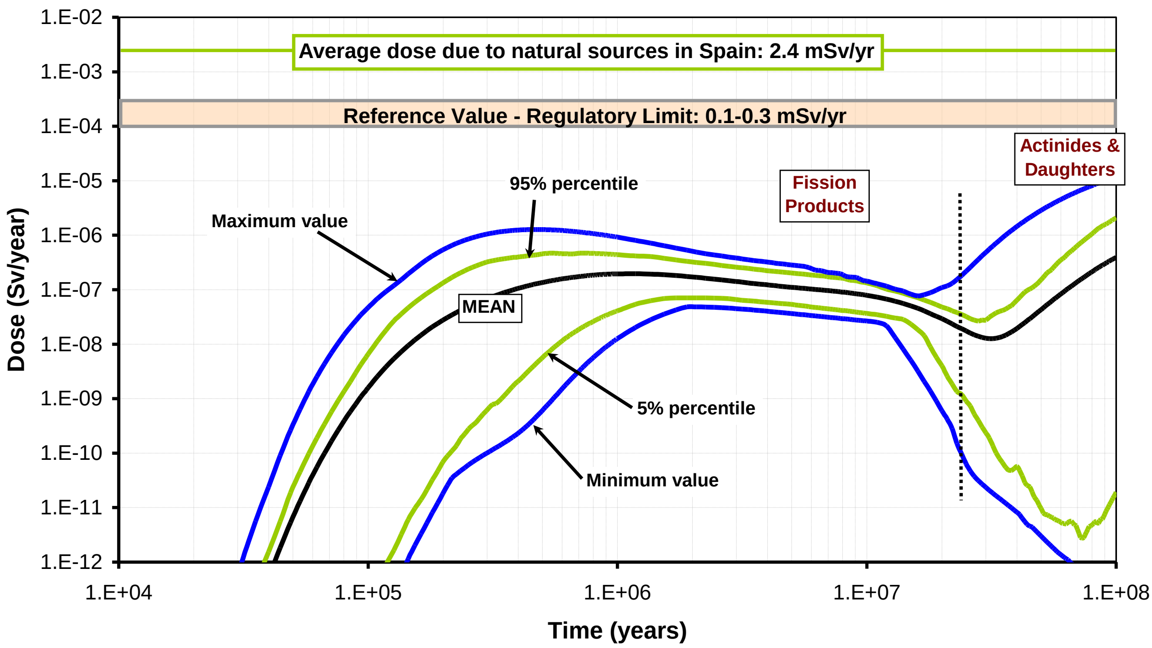

Figure 6.2 Probabilistic calculations for the effective dose rate (ENRESA).

Figure 6.2 presents the results of the probabilistic calculations for the effective dose rate. It illustrates the time evolution of the mean dose, the 5th and 95th percentiles as well as the maximum and minimum doses in a probabilistic calculation (1,000 runs).

In all the runs of the probabilistic calculation the peak dose is at least one order of magnitude below the reference value. This is a useful result, which shows that the existing parameter uncertainty is acceptable from the point of view of fulfilling the dose limits.

6.2 The Dutch concept in clay (NRG)

Figure 6.3 shows the four safety indicators considered in the Dutch concept for the Boom Clay formation normalised by their reference values. Due to the lack of an elaborated model of the geosphere, the power density is calculated for river water and has a different reference value. All indicators lie in a relatively narrow range of about one order of magnitude.

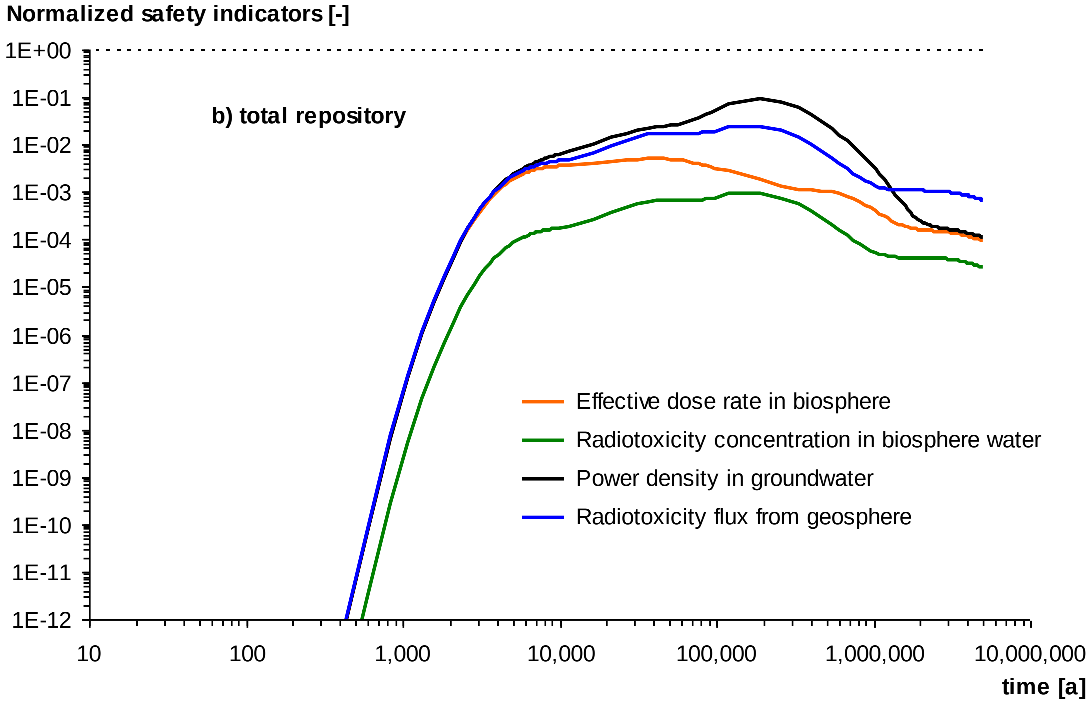

Figure 6.3 The normalised safety indicators (clay) for the total repository calculated by NRG.

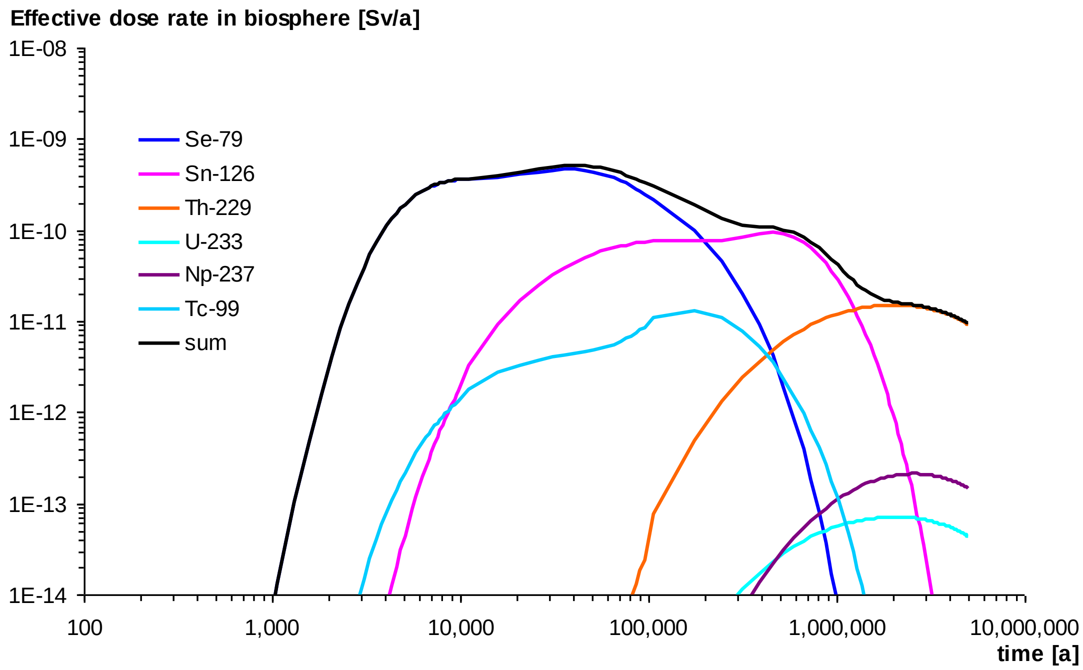

Figure 6.4 The effective dose rate (clay) for the total repository calculated by NRG.

The effective dose rate and the contributions of the most relevant radionuclides to the total effective dose rate are depicted in Figure 6.4. The most relevant radionuclide in the earlier phase of the release to the biosphere is non-sorbing anion Se-79. Compared to Sn-126, the second relevant nuclide, Se-79 is transported three times faster through the geosphere. Th-229, the result of the decay of Pu-241 and Am-241 (and daughter-nuclides), is the most relevant nuclide for later time steps.

6.3 The Belgian concept in clay (SCK·CEN)

For the Belgian concept in the Boom Clay formation the curves of the calculated safety indicators are shown in Figure 6.5. The shapes of the curves are similar, although not identical. The considered safety indicators show that it takes several thousands of years before significant amounts of radionuclides are released from the host formation into the aquifer and eventually into the biosphere.

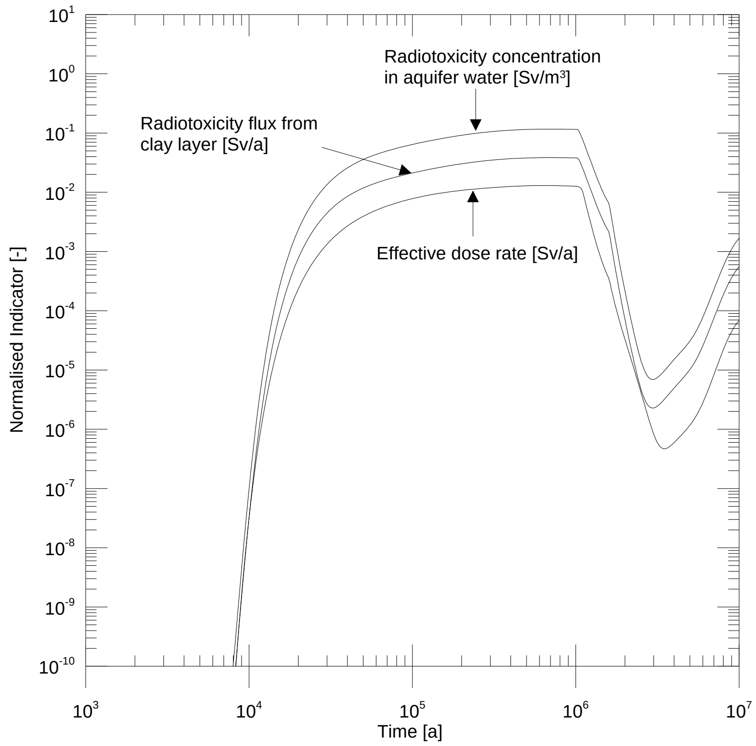

Figure 6.5 The normalised safety indicators calculated by SCK·CEN.

Even if the shapes are very similar the rankings of individual radionuclides as contributors to the total value can be quite different. The reason is that for the dose calculation, the biosphere model considers three exposure paths: ingestion, external exposure and inhalation. For ingestion the use of contaminated water for drinking, irrigation and cattle watering is considered. On the other hand, the calculation of radiotoxicity refers to the ingestion of water. For some radionuclides, such as Se-79, the intake of food (beef and vegetables) gives a high contribution to the dose. For I-129 and most actinides the intake of drinking water is the most important exposure path, whereas for Sn-126 it is external exposure.

Figure 6.6 illustrates the radiotoxicity flux released from the host formation. The calculation of this indicator does not require the results of a biosphere model neither assumptions on the future hydrogeology. The main assumption for the calculation of this indicator is that the transport properties of the host formation will not vary over the considered time scale.

It was not possible to derive a reference value for the radiotoxicity flux from the data available for the candidate host formation. A reference value has been derived from data on the application of fertilizers, which contain various naturally occurring actinides, in Flanders. If one assumes that a typical size of a geological repository that can accommodate the spent fuel arising from the present Belgian nuclear programme is 1 km², a possible reference value for the radiotoxicity flux might be 10 Sv/a.

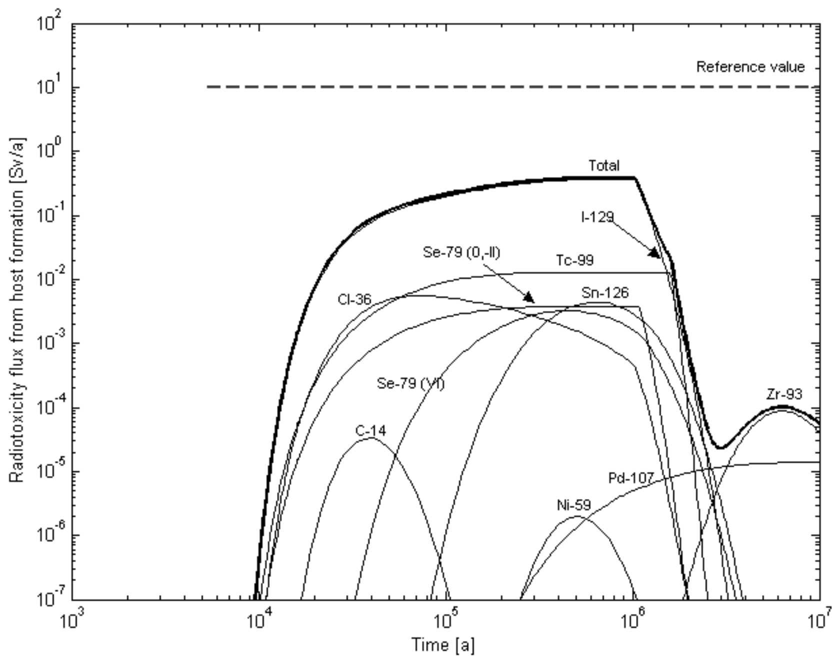

Figure 6.6 Radiotoxicity flux released from the Boom Clay formation calculated by SCK·CEN.

6.4 The German concept in salt (GRS)

All safety indicators calculated by GRS for the German concept in a rock salt formation are normalised by their corresponding reference value (Figure 6.7). In general the temporal evolution of the safety indicators is quite similar. The only exception is radiotoxicity flux from the repository to the geosphere (blue solid line), because this indicator refers to a deeper part of the repository system and is based on fluxes from the repository. Here some radionuclides that are sorbed in the upper groundwater system can play an important role. This indicator was added to show the effect of the geosphere. It anticipates the discussion on repository performance in chapter 7.

The effective dose rate and the radiotoxicity concentration in biosphere water only differ in the radionuclide weighting scheme (the dose conversion factors in case of the effective dose and the ingestion dose coefficients in case of radiotoxicity). Since only a few radionuclides such as I-129, Cs-135, and Cl-36 dominate these quantities, the curves' shapes are quite similar.

Although the reference values are derived independently, the normalised radiotoxicity concentration in biosphere water and the normalised radiotoxicity flux from the geosphere to the biosphere (blue dash and dot line) are almost the identical: If the radiotoxicity flux from the geosphere is divided by the natural groundwater flow the reference value for the radiotoxicity concentration in the geosphere is $2.1 \cdot 10^{-6}$ Sv/m^3 (0.1 Sv/a divided by 48,000 m^3). The reference value for the radiotoxicity concentration in drinking water is $2.0 \cdot 10^{-6}$ Sv/m^3. Since the natural groundwater flux in the calculation is constant, the resulting indicators give almost the same safety margin to the reference value.

The normalised power density in groundwater differs slightly from this overall scheme since Cl-36, the dominating radionuclide at early times is more weighted by the power densities compared with the values of I-129 and Cs-135.

For the given parameter set (chapter 3.4) all indicators remain below their reference values for the calculated period of one million years. The normalised indicators give safety margins between one order of magnitude (radiotoxicity flux into geosphere and power density in upper groundwater) and more than two orders of magnitude (effective dose rate).

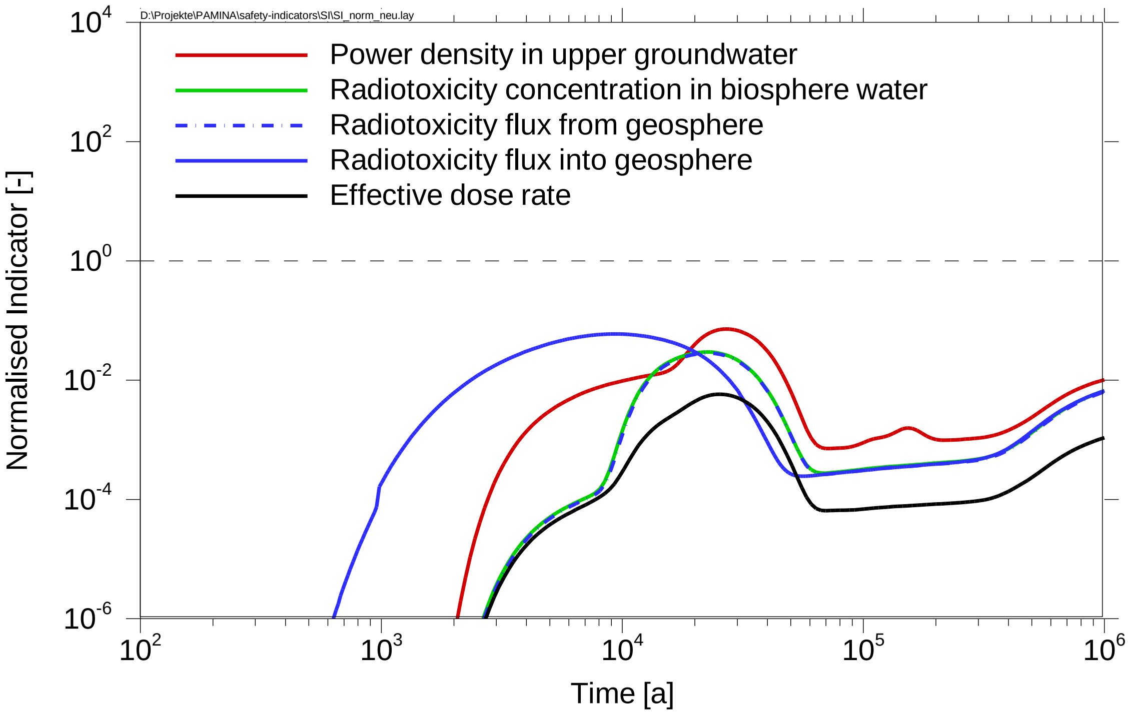

Figure 6.7 The normalised safety indicators calculated by GRS.

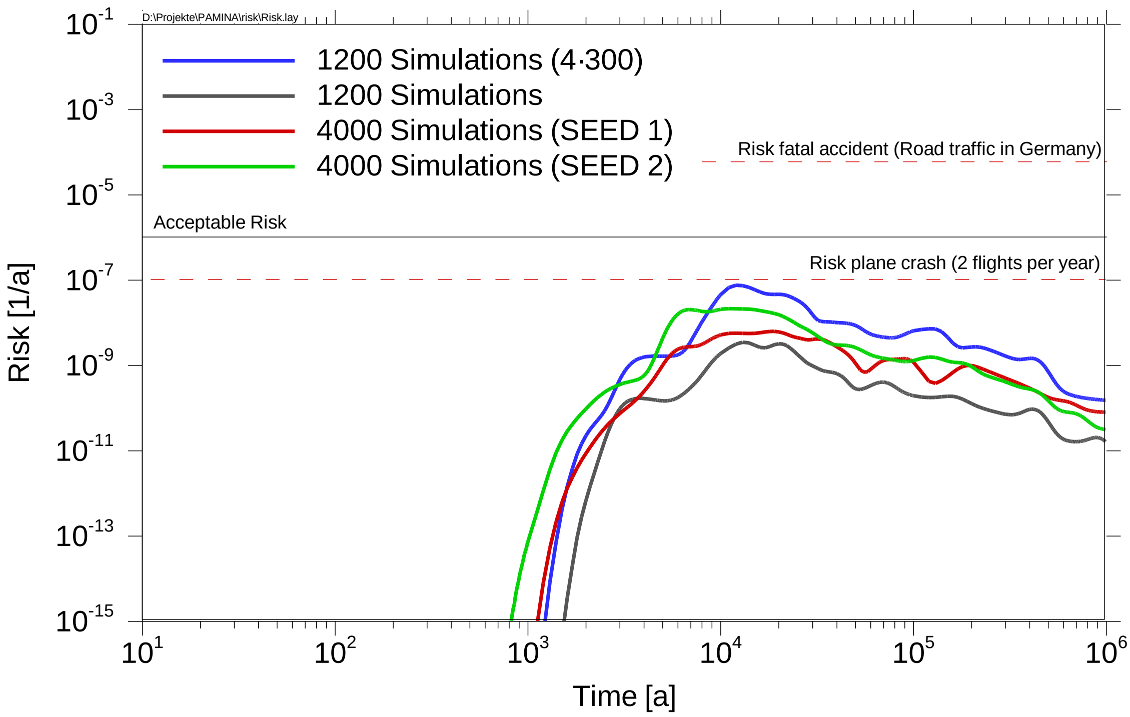

Figure 6.8 The risk calculated by GRS for a repository in salt

Finally, Figure 6.8 depicts the risk calculated for the German concept. Four different MC-simulations were carried out: Two simulations with 1,200 realisations and two simulations with 4,000 realisations and different seeds.

The results in Figure 6.8 show that the risk emanating from the repository system used in this report remains below for all simulations $1 \cdot 10^{-7}$ per year. This is the risk of dying in a plane crash and one order of magnitude lower than the acceptable risk.

6.5 The Dutch concept in salt (NRG)

Figure 6.9 shows for the Dutch concept in salt that all normalised indicators approximately show the same trends and that the normalised values are orders of magnitude below 1.

Figure 6.10 presents the effective dose rate and the most relevant radionuclides. The most relevant radionuclide in the earlier phase of the release to the biosphere is Se-79. Compared to Cs-137, the second most relevant nuclide, Se-79 is transported about three times faster through the geosphere, which eventually results in a higher effective dose rate at an earlier time.

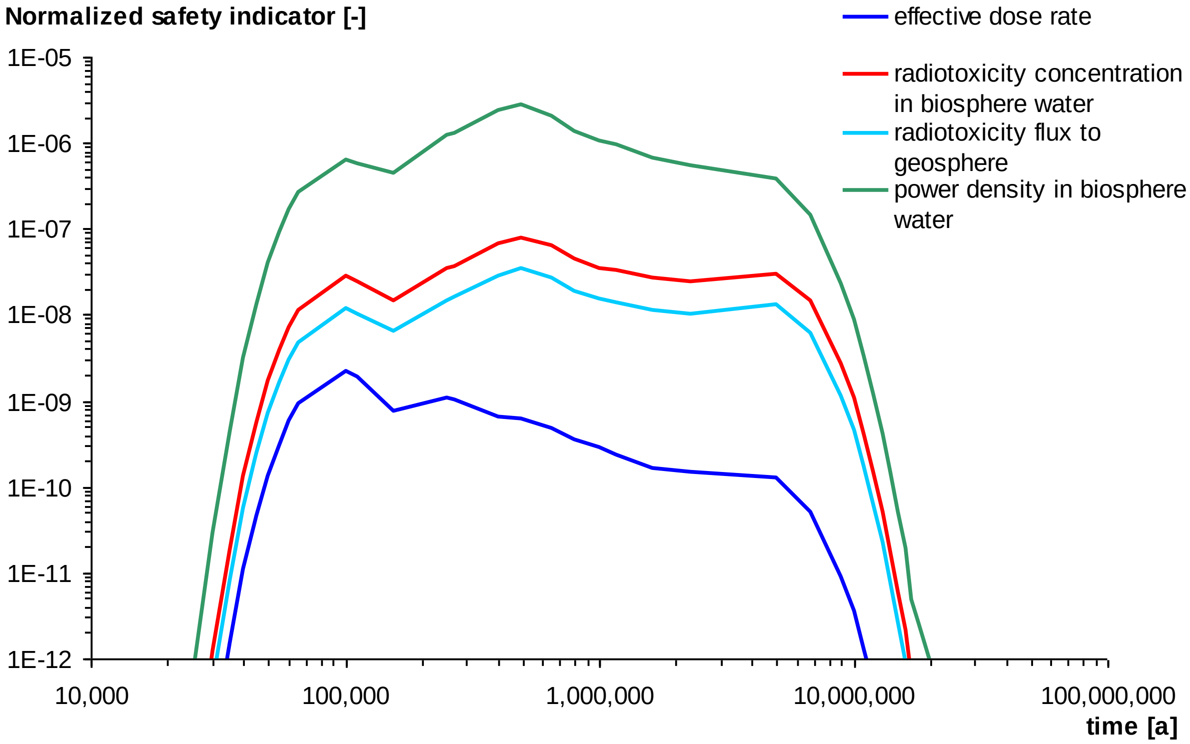

Figure 6.9 The normalised safety indicators (salt) calculated by NRG.

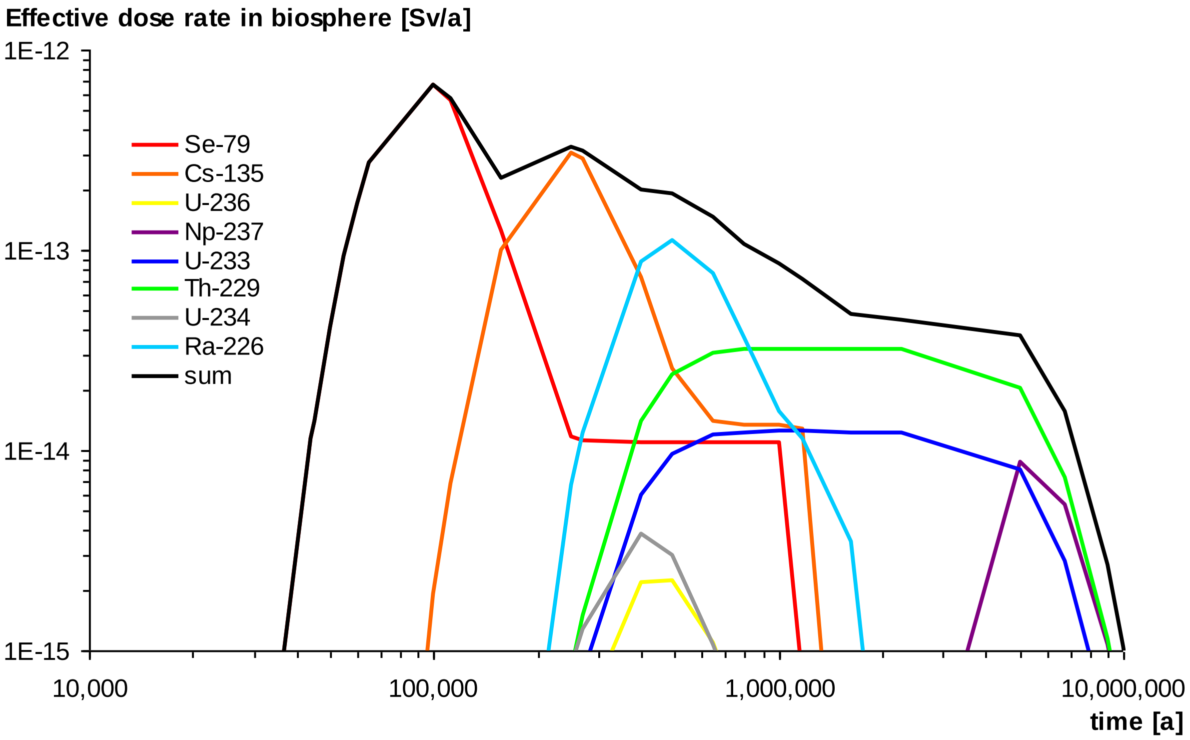

Figure 6.10 Effective dose rate including the contribution of the most relevant nuclides.

At later phases Cs-135 and Ra-226 contribute subsequently for the most part to the total effective dose rate. Ra-226 is formed by the build-up from Th-230 (half-life $7.5 \cdot 10^4$ a) and later on U-234 (half-life $2.5 \cdot 10^5$ a), so its inventory in the geosphere increases in time and peaks at about $1.5 \cdot 10^5$ a (2 half-lives of Th-230, 0.6 half-lives of U-234). In addition, the dose conversion factor of Ra-226 is about 20 times larger than that of U-234, whereas U-234 is transported about 20-fold more slowly than Ra-226 due to differences in sorption properties. This explains the development of the contributions of U-234 and Ra-226.

Note that the behaviour of U-233 is different from that of the other considered uranium isotopes U-234 and U-236. U-233 is formed by the decay of Np-237, which has a relatively large inventory compared to the uranium isotopes. In addition, Neptunium has a larger sorption coefficient and a lower solubility limit compared to uranium. These properties delay the transport of neptunium through the system compared to uranium. During its relatively slow transport through the system Np-237 acts as a “source” of U-233 through its decay. As a result U-233 is present in the system for a much longer time span than U-234 and U-236.

6.6 The Czech concept in granite (NRI)

Finally, the normalised safety indicators calculated by NRI are presented in Figure 6.11. It can be seen that all indicators remain at least one order of magnitude below their reference values.

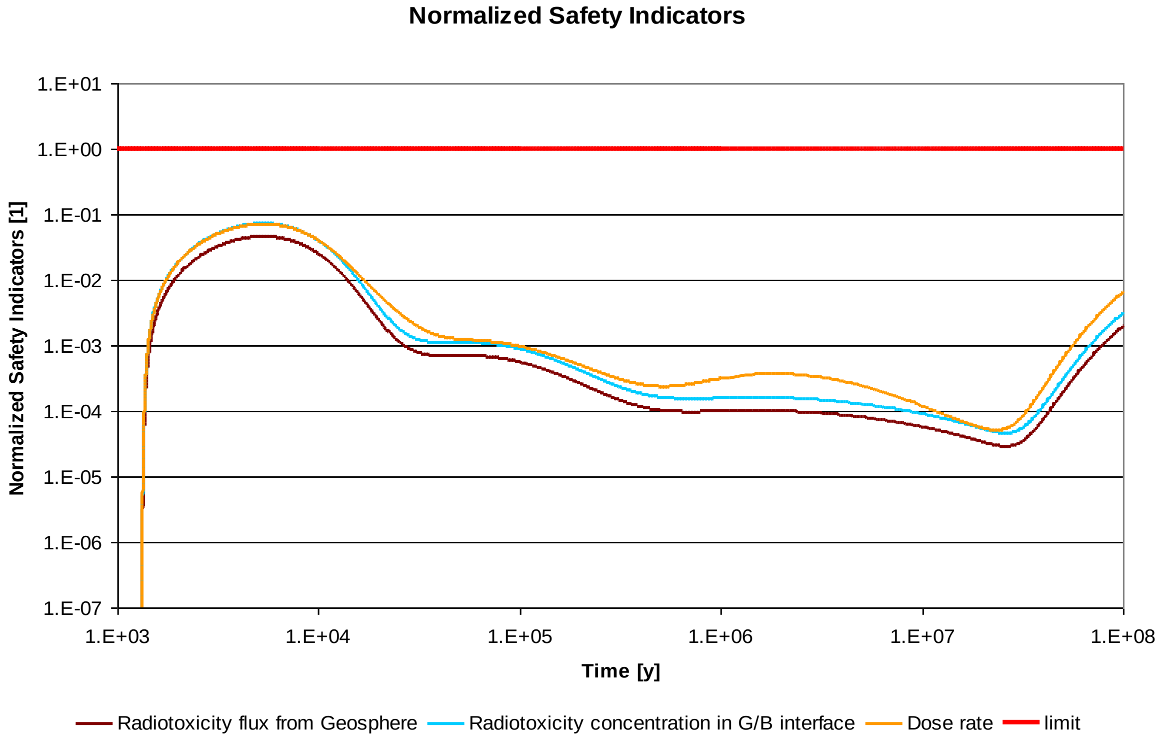

Figure 6.11 The normalised safety indicators calculated by NRI for granite.

The effective dose rate, including the contributions of the most relevant radionuclides are presented in Figure 6.12

The maximal impact of the repository is in early times after the failure of the first canisters due to the release of mobile radionuclides, especially I-129, C-14, Sn-126 and Cl-36. Later the effective dose rate is dominated by the contribution from Cs-135 and Ra-226 (Figure 6.12).

The important factor in the concept of disposal of spent fuel assemblies in carbon steel canisters surrounded by bentonite in granite host rock is primarily the failure rate of canisters. But the results achieved are based on very conservative data, because the Czech deep geological repository programme is still in a conceptual phase.

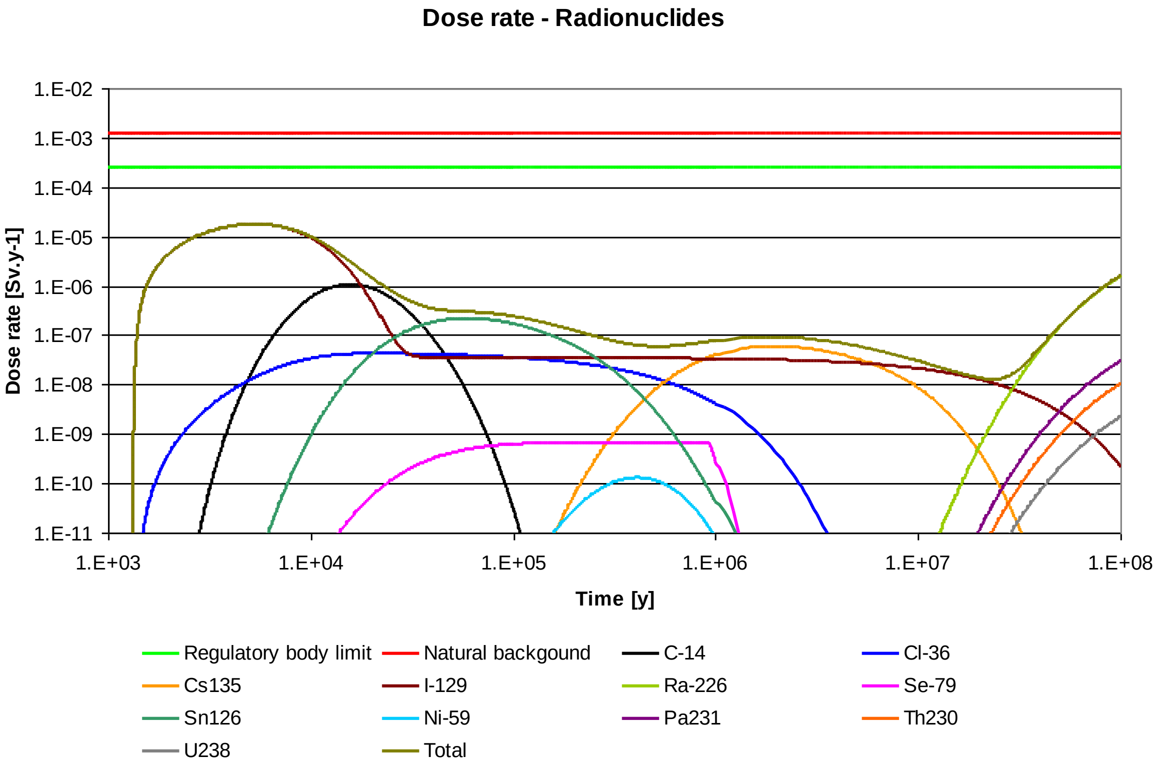

Figure 6.12 The effective dose rate, including the contributions of the most relevant radionuclides, calculated by NRI for granite.

7. Performance indicators

The performance of a repository system is provided by the individual barriers or compartments of the multi-barrier system. The capabilities of the individual barriers to confine radionuclides in the individual compartments and to delay their transport to the biosphere can be quantified using magnitudes such as activities, concentrations and fluxes of radionuclides in and between the successive compartments. Such quantities illustrate the performance of the multi-barrier system and give an indication of the contribution of each considered barrier or compartment to the total performance of the repository. They are therefore considered as performance indicators.

The performance indicators proposed in the SPIN project were aimed mainly at quantifying the contribution of a barrier to the overall performance of the repository system. Since there exists now a trend in safety cases to present the repository as a multi-safety-function system instead of a multi-barrier system, it was also considered desirable to complement the performance indicators with a set of indicators that quantify the contribution of each safety function to the overall performance of the repository system.

The following sections provide an overview of the performance indicators that have been assessed in PAMINA WP 3.4, and an overview of compartments and components of a repository system that may be considered to apply these indicators.

7.1 Indicators based on inventories or concentrations

Performance indicators that are based on inventories or concentrations (Table 7.1) have been thoroughly investigated in the SPIN project; they provide information about:

- Inventories (activities) in compartments, showing where radionuclides are at different points in time,
- Inventories (activities) outside compartments, showing the retention capability of the inner barriers,
- concentrations in compartment water, showing the dilution in successive compartments.

7.2 Indicators based on fluxes

Indicators that relate to fluxes from compartments (Table 7.2) show the evolution of the transport rates of radionuclides between successive compartments, including radioactive decay and ingrowth. They are also a measure of the barrier function of the disposal system.

7.3 Indicators based on integrated fluxes

These indicators can be calculated in absolute terms or normalised to the initial inventory at the repository closure (Table 7.3). They show the retention capabilities of each compartment of the disposal system and are independent of the biosphere model and dilution. If applied to the total sum of inventories, fluxes from inner compartments can be dominated by the contributions from short-lived, highly mobile decay products of high activity fluxes and dose conversion factors. In that case the value of these indicators can exceed the initial inventory. For individual radionuclides these indicators allow the quantification of the fraction of the inventory that decays in each compartment.

Table 7.1 Performance indicators based on inventories or concentrations

Performance indicator	Unit
Activity in compartments	Bq
Activity outside compartments	Bq
Radiotoxicity in compartments	Sv
Radiotoxicity outside compartments	Sv
Activity concentration in compartment water	Bq/m^3
Radiotoxicity concentration in compartment water	Sv/m^3
Concentration in biosphere water / waste package water	-

Table 7.2 Performance indicators based on fluxes

Performance indicator	Unit
Activity flux from compartments	Bq/a
Radiotoxicity flux from compartments	Sv/a

Table 7.3 Performance indicators based on integrated fluxes

Performance indicator	Unit
Time-integrated activity flux from compartments	Bq
Time-integrated radiotoxicity flux from compartments	Sv

7.4 Indicators based on safety functions (SCK·CEN)

In recent years there has been a trend in the development of safety cases to consider a geological repository as a multi-safety-functions system, more than a multi-barrier system. It is therefore considered desirable to complement the performance indicators with a set of indicators that quantifies the contribution of each safety function to the overall performance of the repository system. SCK·CEN defined performance indicators for each of the safety functions identified to contribute to the confinement of the radionuclides in the repository system (Table 7.4):

- containment (PI_C): activity in waste package at t_1 / activity in disposed waste at t_0;
- limitation of release (PI_{R1}): activity released from waste package at t / activity in waste package at t_1;
- retardation (PI_{R3}): activity released from host formation at t / activity released from waste package at t;
- Integrated repository system performance: activity released from host formation at t / activity in disposed waste at t_0; $PI_{IRS} = PI_C \cdot PI_{R1} \cdot PI_{R3}$.

with the disposal time t_0 and the time of overpack (canister) failure t_1

Table 7.4 Performance indicators based on safety functions

Performance indicator	Unit
Containment factor (PI_C)	-
Limitation of release (PI_{R1})	-
Retardation factor (PI_{R3})	-
Performance of the integrated repository system (PI_{IRS})	-

Table 7.5 Performance indicators based on transport times

Performance indicator	Unit
Transport time through compartments	a

7.5 Indicators based on transport times

Comparison of the transport times through compartments with the half-life of a radionuclide provides information about how much will a radionuclide decay during its transit through a compartment and therefore indicates the importance of individual radionuclides for long-term safety.

7.6 Indicators assessing the performance of repository systems in salt formations

7.6.1 Integrity of the geological barrier (BGR)

Rock salt is a potential host rock for the final disposal of radioactive waste. The safety function of the geological barrier and of the technical barriers is to ensure the containment of the waste. This safety function might be jeopardized by processes which affect the integrity of the barriers. Therefore, the safety function approach for a repository for high active waste (HAW) in domal rock salt has to address features, events and processes, which might have an impact on the integrity of the geological barrier. The mechanical conditions that are relevant for the integrity of rock salt are the dilatant state stress below the dilatancy boundary, and the fluid pressure: below the fracturing pressure. After the excavation of openings a reduction of the hydrostatic field occurs and deviatoric stresses become relevant.

7.6.2 Time to plug closure (NRG)

As a result of the naturally-occurring convergence process of salt-based repositories, the compacted salt grit plug that seals off an underlying borehole will experience stress from the surrounding host rock and will therefore be compacted further as time progresses. At a certain point in time the porosity and therefore also the permeability of the plug will have decreased so-far that it will be virtually impossible for brine to pass through the plug. From that time on the disposal cell is sealed off from the adjacent structures and any release of radionuclides from the disposal cell will be terminated. The boundary between “non-closure” and “closure” of the plug is not exactly definable, but in the present analysis a minimum value of the porosity of the sealing plug of 0.03% has been adopted. This corresponds to a calculated permeability of the compacted salt sealing plug of $k = 3 \cdot 10^{-23}$ m^2. At such a low value the plug has become impermeable for any brine flow.

Table 7.6 Units of special performance indicators for salt formations

Performance indicator	Unit
Integrity of the geological barrier	-
Time to plug closure	a

7.7 Table of compartments / components of the repository systems

The general idea of the concept of performance indicators is to look in detail at the transport processes at relevant locations inside the repository system. Comparing the indicators calculated for different locations (compartments) is often very illustrative for demonstrating the functioning of the system.

Compartments can represent natural or mined subsystems like the geosphere or the mine building, engineered components like canisters or barriers, or even physically independent phases in specific regions, like the canister water or the precipitate.

Table 7.7 Compartments considered in WP 3.4 for the different repository systems

Compartment	ENRESA (clay)	SCK·CEN (clay)	NRI (granite)	NRG (clay)	NRG (salt)	GRS (salt)
Waste matrix	X	X	X	X	X	X
Waste package / borehole		X	X	X	X	X
Waste package water / borehole water			X	X	X	X
Precipitate	X	X				X
Buffer	X	X	X	X		
Repository structure					X	X
Host formation	X[(a)]	X	X	X		
Overlying rock	X	X		X	X	X
Biosphere	X	X	X	X	X	X

[(a)] The host rock is divided into five zones

The compartments that are selected in WP 3.4 for the different repository systems are (Table 7.7):

- Waste matrix: It represents the initially emplaced waste taking into account its decay

- Waste package / borehole: The waste container together with its content is called waste package. In the salt concept these compartments comprise the boreholes, where the waste packages are emplaced.

- Waste package water / borehole water: The waste package water (borehole water) refers to all liquid in the waste package that contains radionuclides in a soluble form.

- Precipitate: It includes all radionuclides in the waste package (borehole) that are neither in the waste matrix nor dissolved in the water.

- Buffer (clay and granite): In the clay and granite concepts the waste package is surrounded by a bentonite buffer.

- Repository structure (only salt): Due the impermeability of undisturbed rock salt, a potential transport of contaminants takes place in the disturbed areas of the repository (drift and shaft).

- Host formation (clay and granite): The host formation represents the geological units that belong to the host rock but are located outside of the waste package (and buffer). In the granite concept these units are covered by the geosphere.

- Overlying rock: These compartments represent the geological units between host rock formation and biosphere

- Biosphere: The biosphere is the endpoint of the calculations. It receives the potential contaminant fluxes from the underlying geological units.

8. Performance indicator results

In this chapter the performance indicators that have been calculated by the WP 3.4 participants are presented and assessed. As already mentioned in Chapter 7, six types of performance indicators can be distinguished:

- indicators based on inventories or concentrations (chapter 7.1);
- indicators based on fluxes (chapter 7.2);
- indicators based on integrated fluxes (chapter 7.3);
- indicators based on safety functions (chapter 7.4);
- indicators based on transport times (chapter 7.5);
- indicators assessing the performance of repository systems in salt formations (chapter 7.6).

Indicators of the first four types can be calculated for a single radionuclide (fission or activation products), for a decay chain or for a weighted sum over all radionuclides in terms of radiotoxicity. All indicators aim at illustrating the functioning of the repository system. The third and fourth type of indicators are specifically used for quantifying the contributions of the components or the safety functions of a repository system. Indicators of the fifth type are calculated for chemical elements and compared with the half-lives of the corresponding radionuclides. They are used for repository systems in granite and clay. Indicators of the sixth type are not based on radionuclide transport calculations, but on geomechanical calculations of characteristics of the host salt formation.

8.1 Indicators based on inventories or concentrations

As an example of indicators based on inventories, Figure 8.1 gives the evolution of the Se-79 inventory in the different compartments of a repository in clay (SCK·CEN). Figure 8.1 shows that a considerable fraction of the Se-79 remains during a few tens of thousands of years within the waste matrix. During the first 300,000 years more Se-79 is released from the waste matrix at a faster rate than it can migrate into the buffer, and as a consequence, it precipitates. The precipitate dissolves between 300,000 and 1 million years. After 1.5 million years almost all the Se-79 has left the barrier system and afterwards it decays completely in the biosphere.

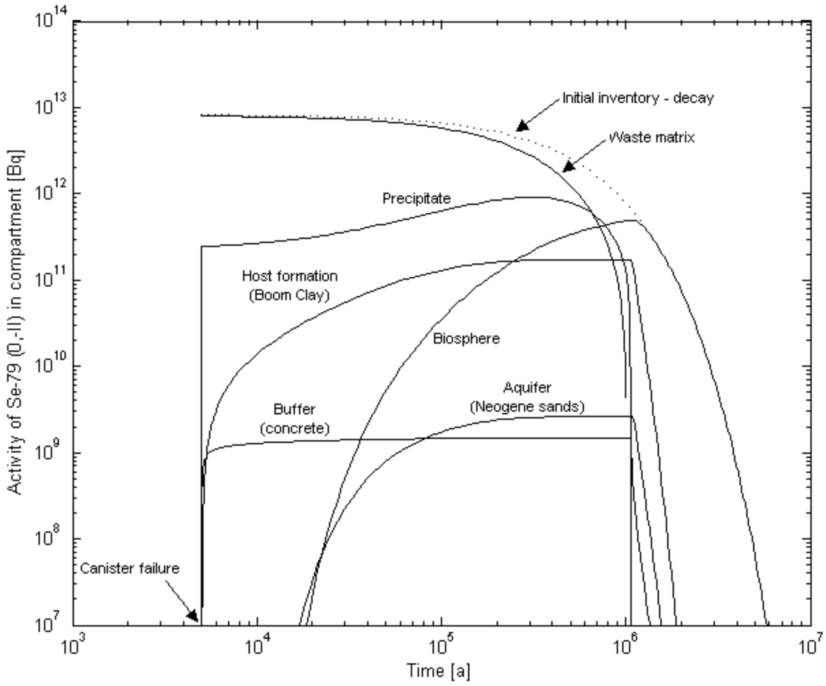

Figure 8.1 Evolution of the activity of Se-79 in the compartments of a repository in clay (SCK·CEN).

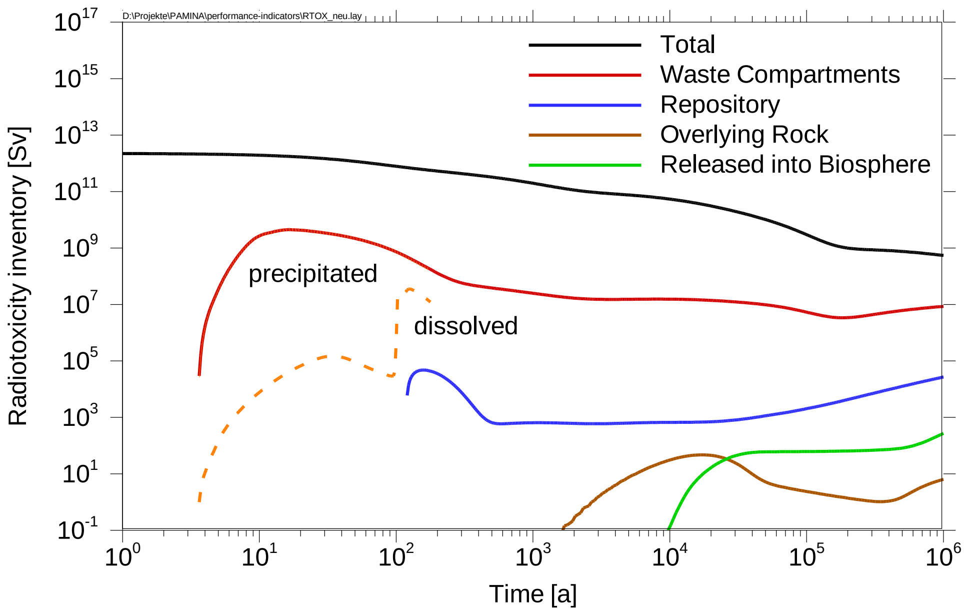

Figure 8.2 Radiotoxicity inventory in different compartments of a repository in salt (GRS).

The evolution of the radiotoxicity inventory in the different compartments of a repository in salt (GRS) for a brine intrusion scenario is shown in Figure 8.2. After a few years the brine reaches the waste canisters in the first waste compartment. At this time the canister failure and radionuclide mobilisation starts. Initially, almost the whole mobilised fraction is precipitated because of the very small brine volume. The inflowing brine volume increases and after about 100 years the borehole is filled with brine. The brine is transported by convergence to the repository compartment. When convergence decreases the volumes of cavities and void spaces to a certain value, the convergence stops and diffusion determines the transport of radionuclides through the repository. This leads to an increase of the radiotoxicity inventory in the repository compartments. After 1000 years radionuclides are released from the repository to the overlying rock and consequently the radiotoxicity inventory in the overlying rock increases. After 10,000 years the radionuclides reach the biosphere.

8.2 Indicators based on fluxes

Figure 8.3 is an example of indicators based on fluxes; it shows the evolution of the I-129 fluxes between the different compartments of a repository in clay (ENRESA). The I-129 present in the instantaneously released fraction (IRF) of the disposed spent fuel causes an initial high flux leaving the canister cavity directly after canister failure. Thereafter the flux released from the canister cavity corresponds to the gradual alteration of the UO_2 matrix. The matrix is completely altered after about 20 million years. The I-129 flux released from the canister cavity is not significantly attenuated by the bentonite buffer. On the other hand, the host clay formation reduces the maximum flux by several orders of magnitude. The Marls formation, which overlies the host clay formation, delays the fluxes for a few tens of thousands of years, but does not attenuate significantly the maximum flux. The jumps observed at 100,000 years are a consequence of the model adopted, in which the bentonite porewater changes from the “initial” to the “long term” composition at that time, leading to a sudden decrease in the matrix alteration rate.

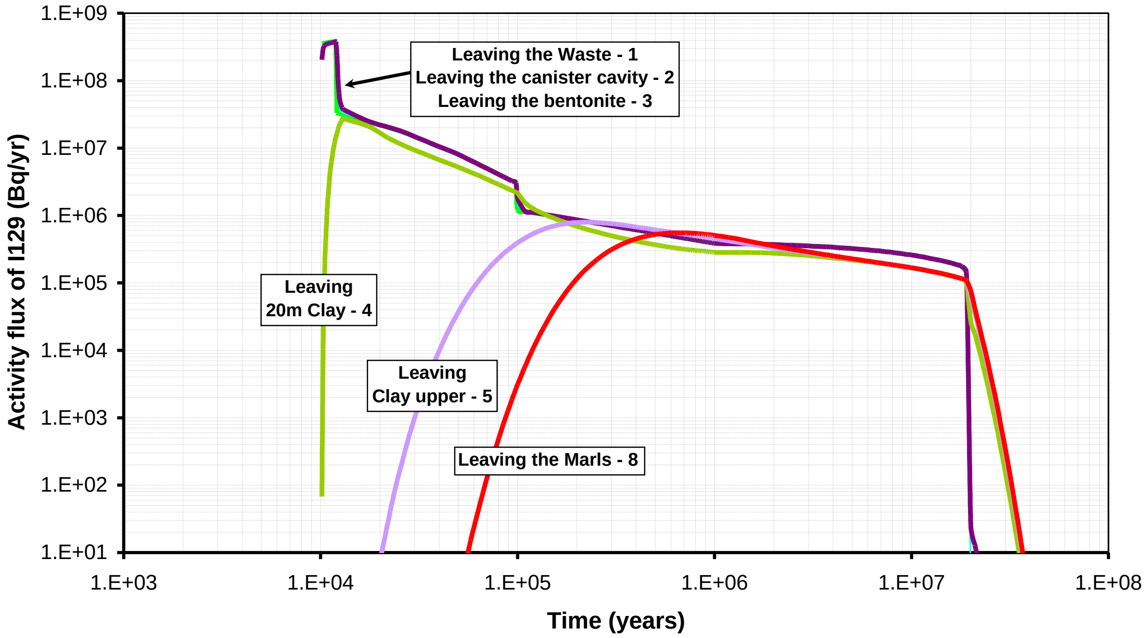

Figure 8.3 I-129 activity fluxes between the compartments of a repository in clay (ENRESA).

Figure 8.4 gives the evolution of the radiotoxicity fluxes from the main compartments of a repository in granite (NRI). Between 1,000 and 100,000 years the radiotoxicity flux released from the waste is strongly reduced by the bentonite buffer. Mobile radionuclides, such as I-129, migrate relatively fast through the granite host formation. On the other hand, for sorbing radionuclides the granite formation gives a significant reduction of the radiotoxicity flux between 20,000 and 100 million years. It is somewhat surprising that the radiotoxicity flux released from the bentonite buffer is after 200,000 years higher than the flux released from the waste; this is explained by the high dose ingestion factors of a number of daughter nuclides in the decay chains.

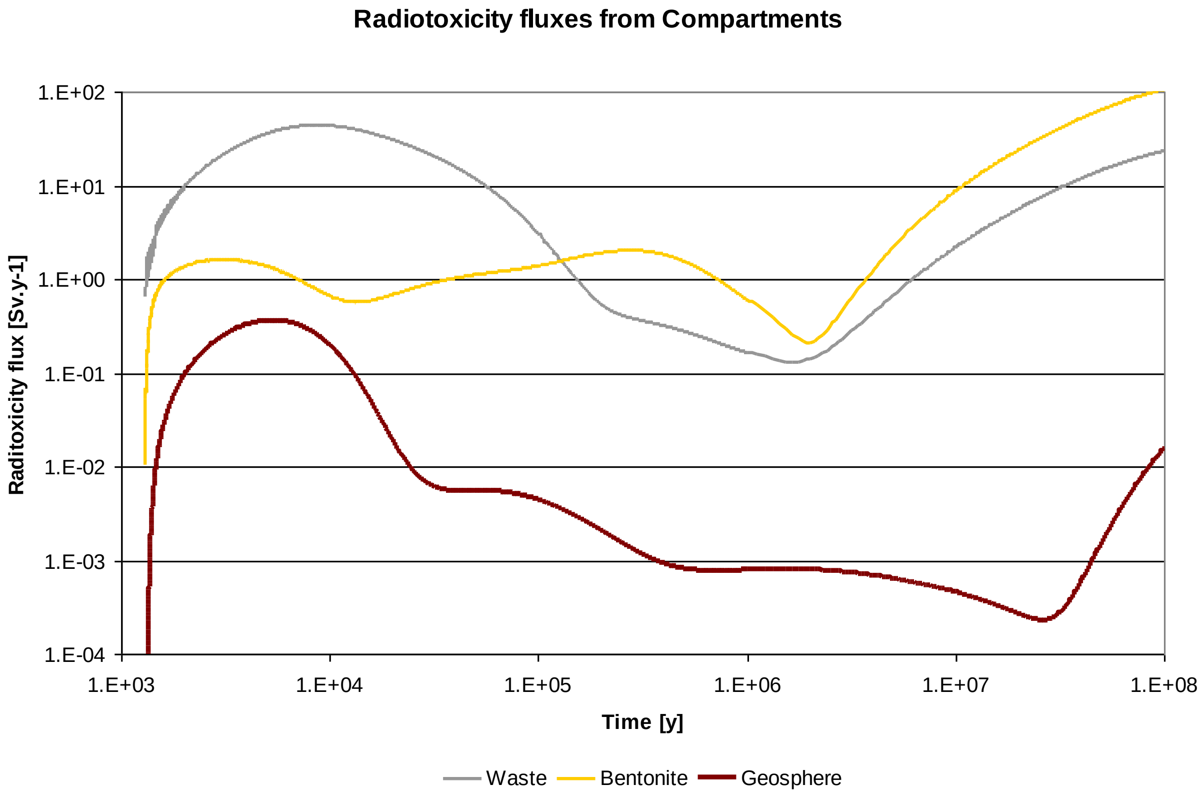

Figure 8.4 Radiotoxicity flux between the compartments of a repository in granite (NRI).

8.3 Indicators based on integrated fluxes

As an example of indicators based on integrated fluxes, Figure 8.5 shows the evolution of the integrated C-14 fluxes released from the compartments of a repository in clay (ENRESA).

The time lag between the different curves illustrates the delay caused by the different barriers. When the curves become horizontal it is possible to calculate the reduction factor provided by a compartment for the considered radionuclide; it is defined as the ratio between the time-integrated fluxes entering and leaving the compartment. In the case of C-14 ($T_{1/2}$ = 5,730 years) it can be seen that the bentonite reduction factor is 1, the 10 m of clay above and below the repository provide a reduction factor of 6, the remaining 90 m of clay above the formation provide a reduction factor of 10,000 and the 110 m of marls provide a reduction factor of 2,000. The product of the reduction factors of the barriers provides the reduction factor of the whole disposal system that in the case of C-14 is 10^8.

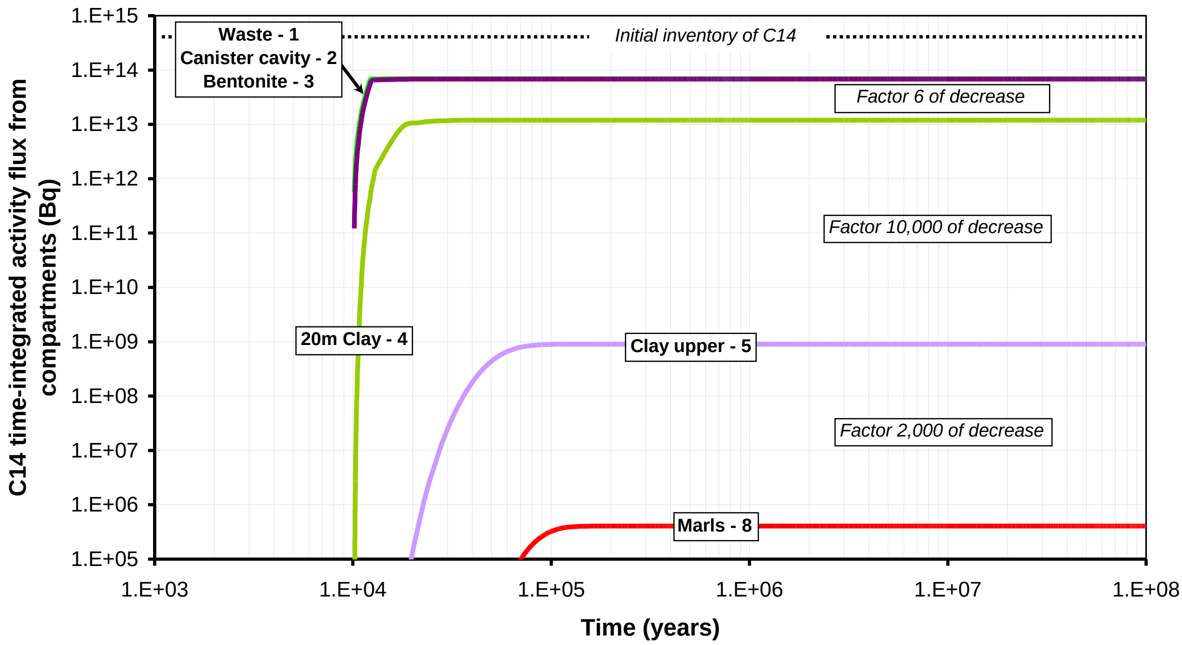

Figure 8.5 Radiotoxicity flux between the compartments of a repository for C-14 in clay (ENRESA).

The time integrated radiotoxicity flux due to fission and activation products calculated by SCK·CEN for a repository in clay is shown in Figure 8.6. The flat curves in Figure 8.6 after about 1.3 million years illustrate that the integrated radiotoxicity due to fission and activation products is completely dominated by the mobile (i.e. R = 1) radionuclides, which have all diffused out of the host clay layer at that time.

The radiotoxicity due to fission and activation products decreased from $2.5 \cdot 10^{11}$ Sv at disposal time to $4.9 \cdot 10^{6}$ Sv at the time of canister failure. The further reduction provided by the waste matrix and solubility limits is limited to a factor 2.5 (Figure 8.6). As the radiotoxicity that reached the biosphere is essentially due to mobile fission and activation products (mainly I-129 and Se-79), the contribution of the host clay formation to the reduction of the radiotoxicity is limited to a factor 2.4.

As the discussion of the time-integrated activity/radiotoxicity fluxes is often strongly focusing upon their final values, Table 8.1 gives the values of the relative (i.e. divided by their initial inventory) time integrated (over a period of 10 million years) activity fluxes calculated for the fission and activation products for a repository in clay (SCK·CEN).

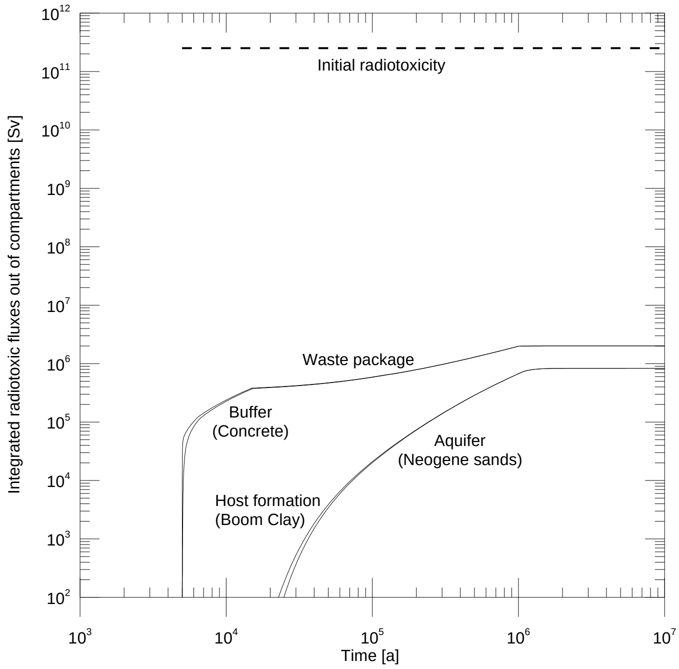

Figure 8.6 Time-integrated radiotoxicity flux due to fission and activation products out of compartments of the disposal system for a repository in clay (SCK·CEN).

Table 8.1 Relative time-integrated (over a 10 million years period) activity fluxes calculated for the fission and activation products for a repository in clay (SCK·CEN)

Radionuclides	Waste package	Buffer	Clay
C-14	6.38E-02	5.58E-02	1.09E-05
Cl-36	4.98E-01	4.96E-01	3.44E-01
Ni-59	8.76E-01	8.70E-01	4.04E-05
Se-79 (VI)	3.98E-01	3.96E-01	1.14E-01
Se-79 (0,-II)	2.24E-01	2.24E-01	1.71E-01
Zr-93	8.14E-01	8.12E-01	1.62E-03
Nb-94	6.82E-01	6.74E-01	1.84E-09
Tc-99	2.96E-02	2.92E-02	2.04E-02
Pd-107	3.64E-01	3.62E-01	2.24E-01
Sn-126	3.04E-01	3.04E-01	1.02E-02
I-129	9.78E-01	9.78E-01	9.72E-01
Cs-135	8.68E-01	8.66E-01	5.78E-08

Table 8.1 shows that the waste package gives a significant contribution for C-14 and Tc-99 and to a lesser extent for Se-79, Cl-36, Pd-107 and Sn-126. For C-14 this contribution is essentially due to its relatively short half-life. For Tc-99 and Pd-107 it is their low solubility that contributes significantly to their retardation within the waste package. As sorption on the

concrete buffer has been conservatively neglected in the performed calculations, the contribution of the buffer is negligible for most radionuclides; only for C-14 it gives a significant contribution because of the short half-life of this radionuclide. Several radionuclides decay to low activity levels during their migration through the host clay layer; this is the case for Nb-94, Cs-135, C-14 and Ni-59, and to a lesser extent for Zr-93 and Sn-126.

The time-integrated radiotoxicity flux after 10 million years (or 1 million years) can be used to calculate the containment factor. This is the radiotoxicity released from the host formation divided by the radiotoxicity inventory at disposal time. This indicator gives a good indication of the confinement provided by the integrated repository system. The containment factors calculated for the fission and activation products for a 10 million years period can be found in the last column of Table 8.1. The smaller the value, the better is the performance of the integrated repository system for a certain radionuclide. The two extremes among the considered radionuclides are Nb-94, which decays almost completely during its migration through the host clay layer, and I-129, of which nearly the whole inventory is released over a long period from the host clay formation.

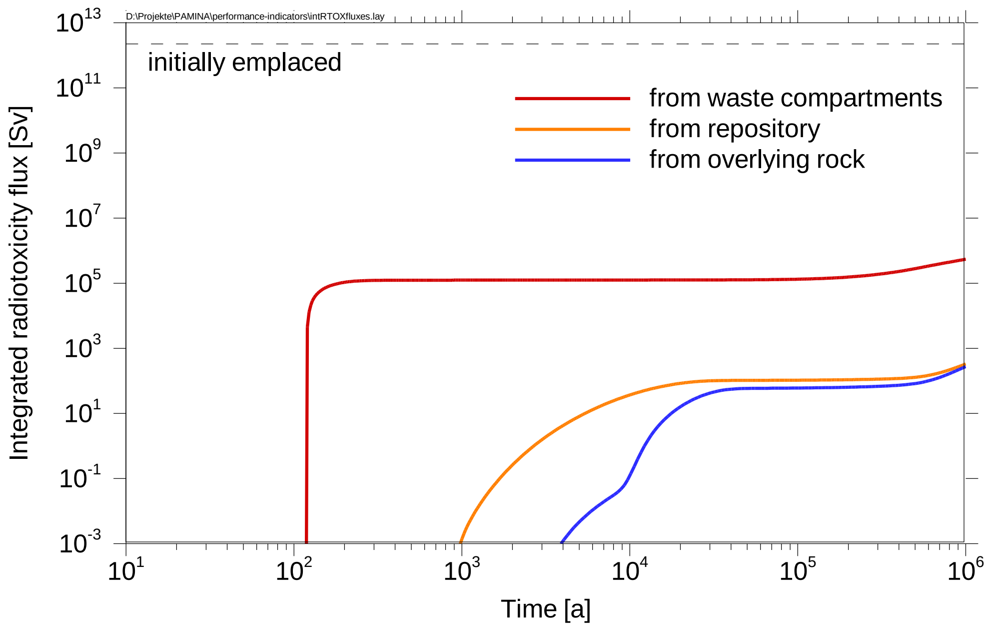

Figure 8.7 Integrated radiotoxicity fluxes from different compartments for all radionuclides calculated for a repository in salt (GRS).

The time-integrated radiotoxicity flux due to all radionuclides calculated for a repository in salt (GRS) is shown in Figure 8.7. The overall initially emplaced radiotoxicity inventory is $2.25 \cdot 10^{12}$ Sv. In one million years the total release of radiotoxicity from the waste compartment is $5.45 \cdot 10^{5}$ Sv, $3.31 \cdot 10^{2}$ Sv from the repository and $2.75 \cdot 10^{2}$ Sv from the overlying rock. The total reduction of the radiotoxicity in the whole repository system

(= containment factor) is about ten orders of magnitude, seven are contributed by the waste compartments and three by the barriers of the repository system.

Figure 8.8 shows the time-integrated radiotoxicity flux from compartments for the whole inventory of radionuclides in the Spanish repository in clay. A surprising result is that, after 200,000 years, the time-integrated radiotoxicity flux from the canister cavity to the bentonite becomes greater than the time-integrated radiotoxicity flux from the waste. This result contradicts the intuitive idea that the amount of radioactive material that leaves a barrier should be smaller than the amount that has entered the barrier. This result is related to radioactive series: most U-238 released from the waste precipitates in the canister cavity, where short lived and more mobile radionuclides (Ra-226) are produced and quickly migrate into the bentonite.

In addition, the time-integrated radiotoxicity flux from compartments can reach very high values, up to four several orders of magnitude higher than the radiotoxicity which is really outside the compartment at the same instant. These very high values ($>10^{12}$ Sv after 10^8 years) can create some confusion and it is recommended to use this indicator with care.

The previous problems with the time-integrated radiotoxicity fluxes can be avoided presenting the results in terms of time-integrated mass (moles) fluxes (Figure 8.9). In this case the time-integrated fluxes behave as intuitively expected, and the flux leaving a compartment is always equal or smaller than the flux entering it.

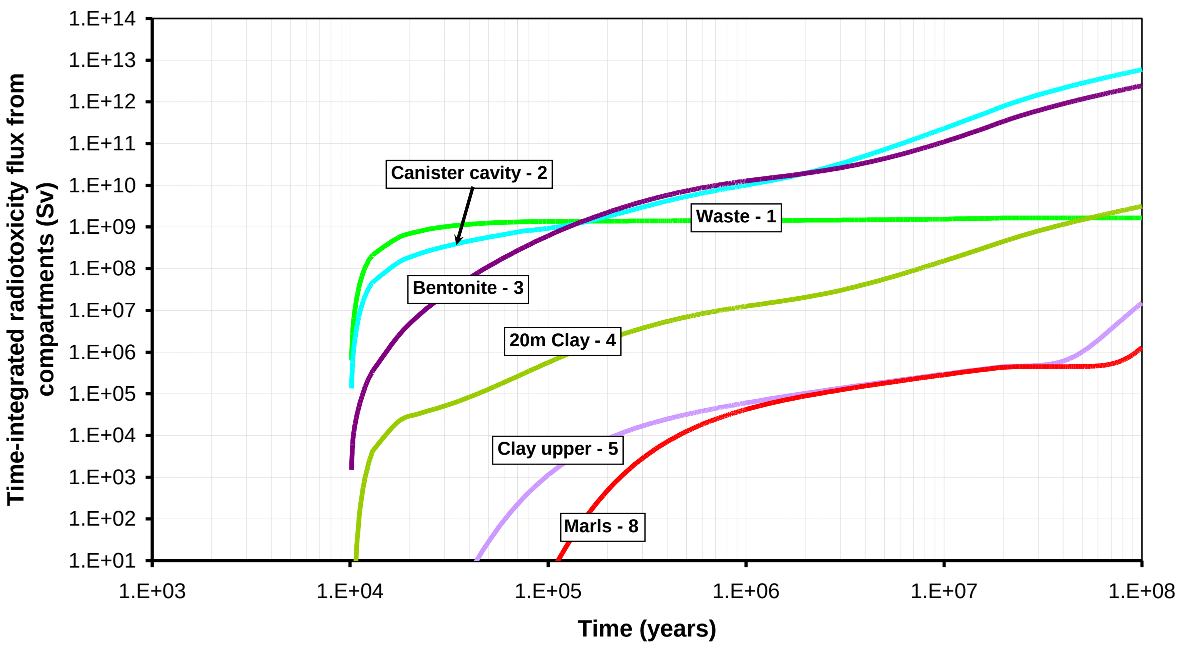

Figure 8.8 Time-integrated radiotoxicity flux from compartments (ENRESA).

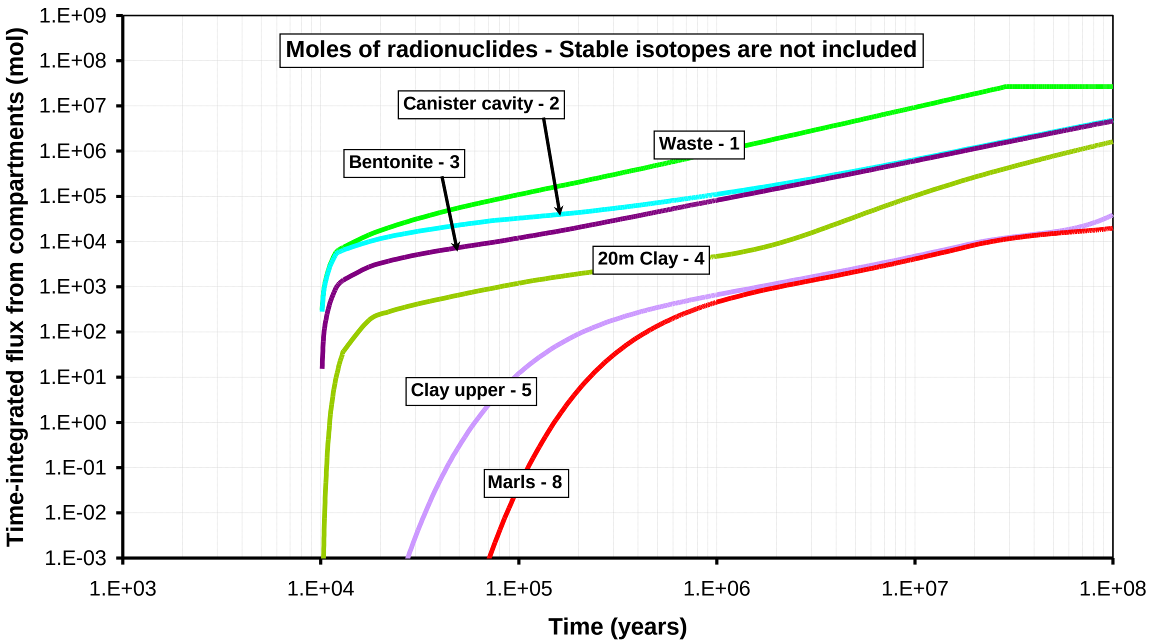

Figure 8.9 Time-integrated mass flux of radionuclides from compartments (ENRESA).

8.4 Indicators based on safety functions

The performance indicators discussed in the three previous sections were based on the compartments of the disposal system. Within PAMINA, SCK·CEN has proposed a new set of performance indicators quantifying the contributions of the active safety functions of the repository system (chapter 7.4).

Figure 8.10 shows the evolution of the performance indicators based on safety functions calculated for Se-79 for a repository in clay (ENRESA). The indicator for the safety function "containment" corresponds to the fraction that decays during the time that the container is intact. The indicator "limitation of release" corresponds to the fraction that is released from the waste package and the indicator "retardation" shows the fraction that is released to the biosphere. The product of these three indicators gives the fraction of the inventory at disposal time that is released from the integrated repository system. In the case of Se-79 the "retardation" safety function gives a contribution that is somewhat larger than the one of the "limitation of release" safety function.

Table 8.2 gives the values of the performance indicators based on safety functions at the end of the considered time scale for a repository in clay (SCK·CEN). It shows that of the considered fission and activation products only C-14 has undergone a significant decay during the containment period. For I-129, none of the three safety functions is effective in reducing its activity, because of its long half-life and high mobility. For Cl-36 the safety function "limitation of release" (provided by the waste matrix) gives a significant contribution to the confinement.

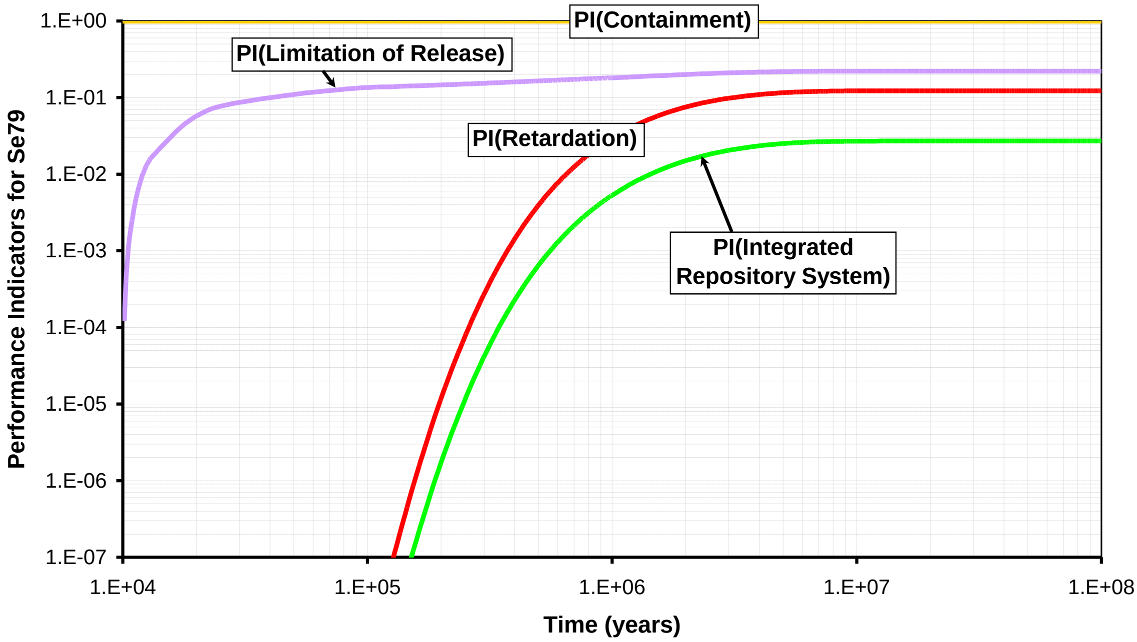

Figure 8.10 Performance indicators related to safety functions for Se-79 for a repository in clay (ENRESA).

Table 8.2 Performance indicators of the contribution of each safety function calculated for a 10 million years period for a repository in clay (SCK·CEN) (C: containment; R1: limitation of release; R3: retardation; IRS: integrated repository system)

Radionuclides	PI_C	PI_{R1}	PI_{R3}	PI_{IRS}
C-14	5.46E-01	3.48E-02	1.70E-04	3.24E-06
Cl-36	9.89E-01	4.93E-01	6.92E-01	3.37E-01
Ni-59	9.55E-01	8.36E-01	4.61E-05	3.68E-05
Se-79 (VI)	9.88E-01	3.94E-01	2.86E-01	1.12E-01
Se-79 (0,-II)	9.88E-01	2.21E-01	7.62E-01	1.67E-01
Zr-93	9.98E-01	8.12E-01	1.99E-03	1.61E-03
Nb-94	8.43E-01	5.74E-01	2.70E-09	1.31E-09
Tc-99	9.84E-01	2.91E-02	6.88E-01	1.97E-02
Pd-107	9.99E-01	3.63E-01	6.20E-01	2.25E-01
Sn-126	9.83E-01	3.00E-01	3.35E-02	9.88E-03
I-129	1.00E+00	9.79E-01	9.93E-01	9.72E-01
Cs-135	9.98E-01	8.66E-01	6.65E-08	5.75E-08

The performance indicators quantifying the safety functions can be calculated for the whole inventory by using radiotoxicity as weighting factor. The resulting time-dependent indicators are shown in Figure 8.11. PI(LR), PI(R) and PI(IRS) do not reach "a plateau" in 100 million years and PI(R) becomes greater than 1 after 3 million years (red region in Figure 8.11). The surprising result observed for PI(LR) is caused by the U-238 precipitated in the canister cavity, which produces short-lived and more mobile daughters that quickly migrate into the bentonite.

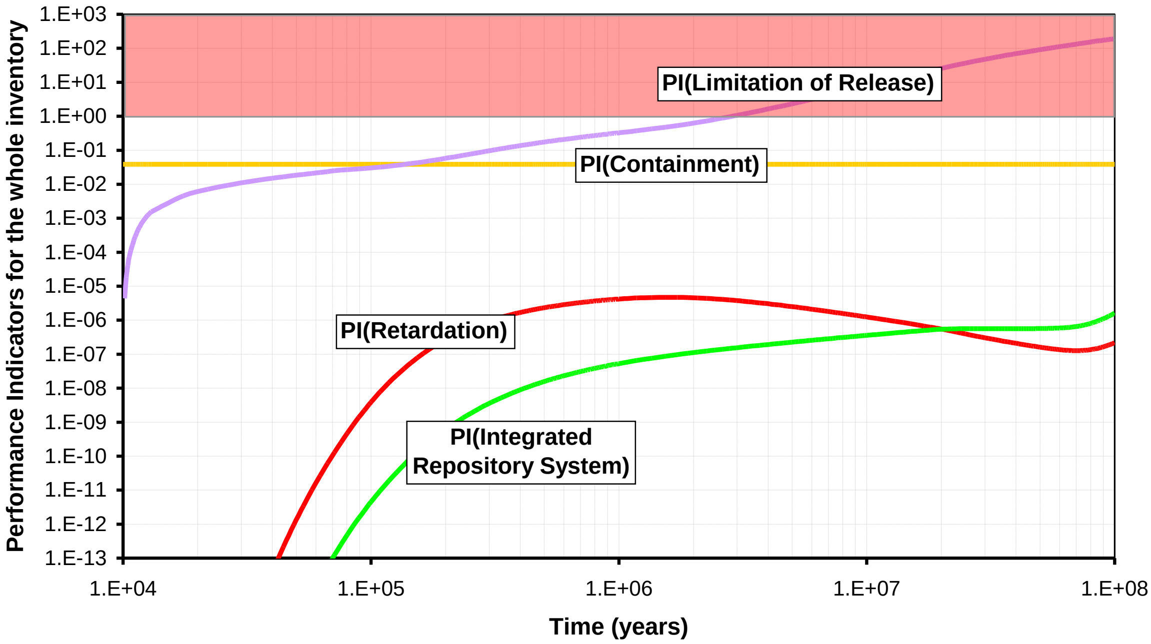

Figure 8.11 Performance Indicators related to safety functions for the whole inventory calculated for a repository in clay by using radiotoxicity as weighting factor (ENRESA).

Table 8.3 Performance indicators of the contribution of each safety function for actinides calculated for a repository in clay by using molar activity as weighting factor (SCK·CEN).

Decay series	PI_C	PI_{R1}	PI_{R3}	PI_{IRS}
4N	9.99E-01	9.39E-02	1.31E-02	1.23E-03
4N+1	1.00E+00	1.43E-01	6.55E-05	9.34E-06
4N+2	1.00E+00	4.83E-04	2.97E-02	1.43E-05
4N+3	1.00E+00	2.33E-02	3.66E-02	8.53E-04

The problem of performance indicators becoming higher than 1 can be solved by using molar activities as weighting factor instead of radiotoxicity. Table 8.3 gives the performance indicators related to safety functions calculated by SCK·CEN after 10 million years for the actinides by using molar activities as weighting factor. All the values in Table 8.3 are equal or smaller than one.

8.5 Indicators based on transport times (granite and clay formations)

In order to explain the functioning of a transport barrier it is useful to define a “transport time through the barrier” that is specific of each chemical element. The definition of this transport time is not unique, and it should be consistent with the intuitive meaning of such concept: “if radionuclide half-life is much smaller than its transport time through the barrier, only a very small fraction of the radionuclide flux that enters the barrier will cross through”.

For a repository in a clay layer a typical diffusion time can be defined as L^2/D_a, where L is the thickness of formation above the repository and D_a is the apparent diffusion coefficient. For an unit instant injection of a solute at repository depth, assuming a thickness L of clay above and below the repository and zero concentration on top and bottom of the formation, the time-integrate flux of solute through the upper surface of the clay is shown in Figure 8.12 for stable and radioactive solutes. In can be seen that for half-lives shorter than $0.01 \cdot L^2/D_a$ the releases from the host formation are negligible.

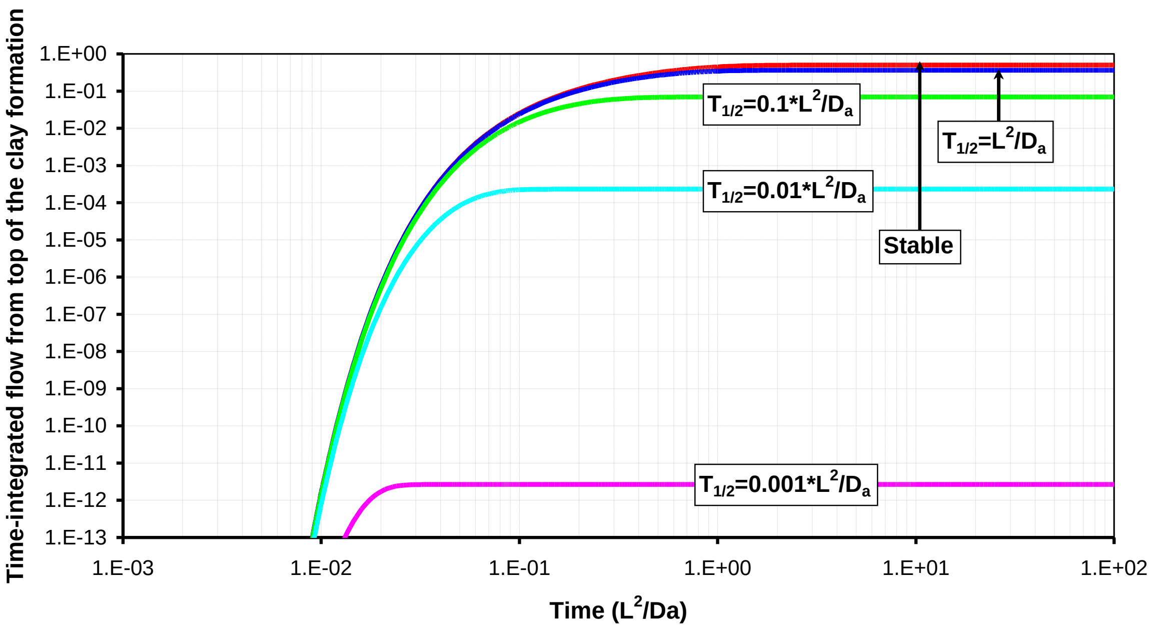

Figure 8.12 Time-integrated fluxes from the top of a clay formation for a unit instant injection of stable and radioactive solutes at repository depth.

ENRESA has found it useful to define a travel time as $0.1 \cdot L^2/D_a$ and to present those travel times together with the radionuclide half-lives (Figure 8.13). This figure allows to present the deterministic values and the probabilistic ranges of the travel times for many radionuclides simultaneously. The green and orange regions clearly show which radionuclides are expected to fully decay in the clay formation and which radionuclide will suffer negligible decay in the formation, respectively

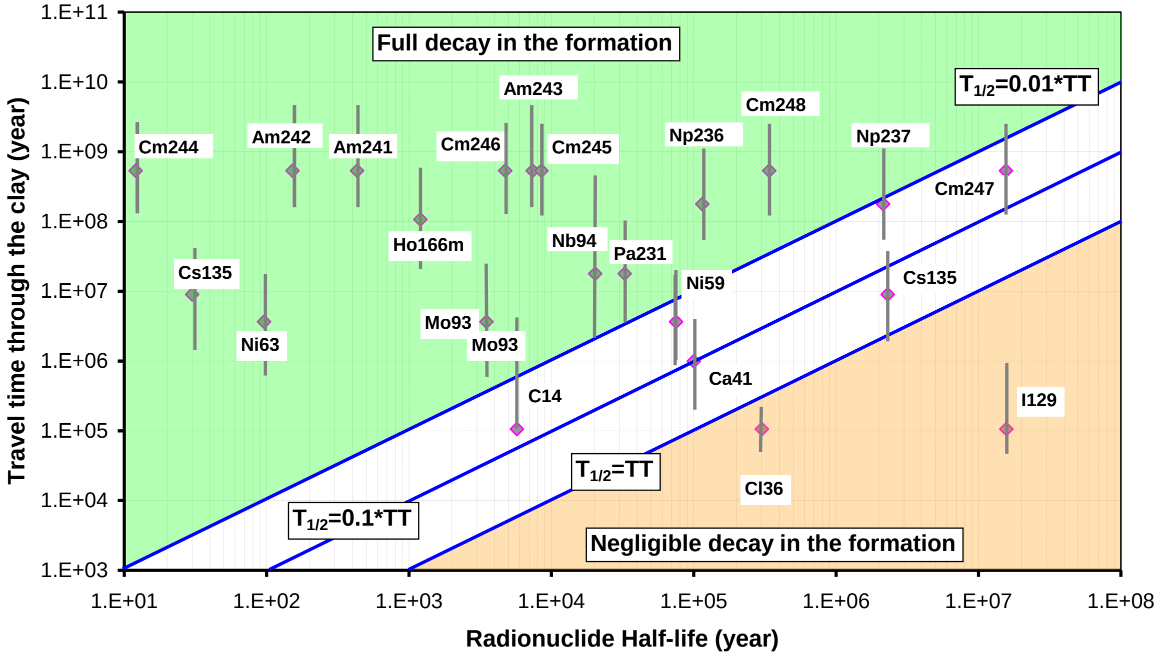

Figure 8.13 Travel times through the clay formation vs. half-lives in deterministic (diamonds) and probabilistic (bars) calculations.

8.6 Indicators based on characteristics of the salt host formation

Two indicators in this last group of performance indicators have been considered for a repository in salt. BGR has calculated indicators based on dilatant state of stress and on fluid pressure. NRG has calculated the time to closure of a compacted salt grit plug that seals off a disposal borehole.

8.6.1 Integrity of the geological salt barrier (BGR)

The performance indicators "dilatant state stress" and "fluid pressure" are analysed by means of the dilatancy boundary value and depth related brine pressure respectively. For these analyses numerical model calculations have been performed. The modelling includes the specific repository site and its geological structure. The modelling results will illustrate the spatial variation of the dilatancy and the parts of the barrier that are prone to fracture risk during the history of the repository and thus provide valuable arguments to strengthen confidence in the safety of the repository system.

For the geomechanical model the geological structure of the host rock and the overburden as well as the geometry of the rooms are idealized and simplified. Figure 8.14 (left) shows the main units of the Zechstein strata (Leine - and Staßfurth formation) and composites of the main units (z3LS-OS-BK/BD) as well as the idealized overburden of the modelled domain. Figure 8.14 (right) illustrates the spatial discretisation of the finite-element-model.

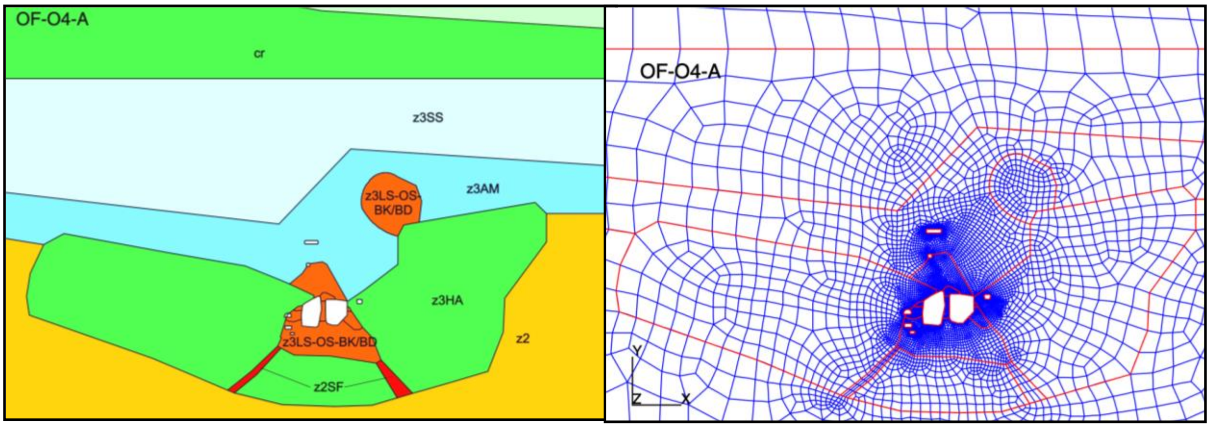

Figure 8.14 Idealized geomechanical model "OF-O4-A" (left) and part of the finite element mesh (right) (z3 layers belong to the Leine formation (z3HA = anhydrite layers) and z2 layers belong to the Staßfurth formation; cr = overburden).

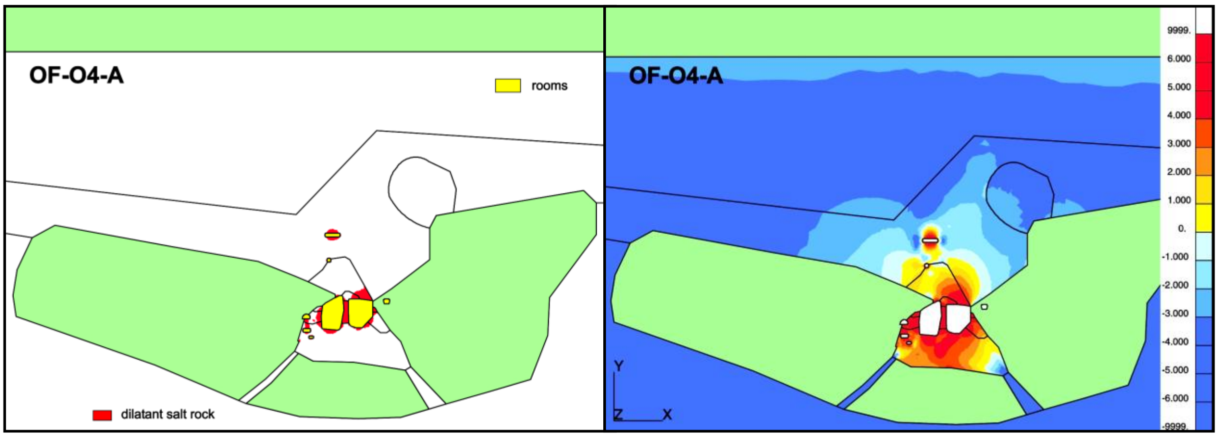

Figure 8.15 Dilatant rock zones (left) and hypothetical rock zones affected hydraulically (right).

For the assessment of the indicator "dilatant state stress", numerical model calculations are performed. If the state of stress does not exceed the dilatancy boundary value, damages do not appear over the long term. Only if the rock stress exceeds the dilatancy boundary value, micro-cracks occur and will cause progressive damage of the barrier integrity. In general the modelling results show that dilatant rock zones basically arise around the mining rooms as a consequence of excavation effects and creep of the salt rock. Figure 8.15 (left) illustrates predicted dilatant zones in rock salt 100 years after excavation of the rooms. In some parts the dilatant damaged zones in the near field of the openings (galleries and disposal rooms) expand to the anhydrite layers in their vicinity. Large parts of the salt barrier between the disposal rooms and the top of the salt dome show no dilatancy. From the model results, the development of dilatant rock zones with time seems to be very slow. From laboratory and insitu testing, it is known, that the integrity of the barrier is guaranteed if the hydrostatic pressure does not exceed the minimum principal rock stress at the given location. The hydrostatic pressure results of a hypothetical column of brine which extends to the ground surface. Figure 8.15 (right) depicts the calculated difference between the theoretical maximum brine pressure according to depth and the minimum rock stress in the salt rock 100 years after commencement of the excavation. The calculations yielded zones around the rooms with a distinct reduction of the minimum principal stress. The areas that are hypothetically jeopardized by fracturing due to brine pressure are yellow, orange and red coloured. From a hypothetical point of view, the stress conditions in this area appear to be unfavourable because the brine pressure exceeds the minimum principal stress in the salt rock between the rooms and the anhydrite blocks. Since the anhydrite blocks have no hydraulic connection to the overburden and favourable stress values occur at the top of the salt barrier, a loss of the barrier effectiveness is not to be expected.

For the given example (Figure 8.15) the modelling results of both indicators show no hydraulic connection to the overburden. Furthermore, after 100 years large parts of the salt barrier show neither an exceedance of the dilatancy boundary value nor a hypothetical exposure to brine induced fracturing. Thus, the integrity of the salt barrier is preserved.

8.6.2 Time to closure of a salt plug (NRG)

The “time to closure” is a measure of the sealing of a disposal cell from the adjacent structures after which any release of radionuclides from the disposal cell will be terminated. The boundary between “non-closure” and “closure” of the plug is not exactly definable, but NRG assumed a minimum value of the porosity of the sealing plug of 0.03%, which corresponds to a calculated permeability of the compacted salt sealing plug of $k = 3 \cdot 10^{-23}$ m^2. At such a low value the plug has become impermeable for any brine flow.

As a result of the compaction of the plug the porosity decreases gradually in time (see Figure 8.16) until the calculation is terminated at the pre-determined value of the “closure” porosity. The corresponding average “time of closure” is 354 years (minimum: 233 years, maximum 615 years). The results of the present analyses show that, depending on the scenario and model assumptions, it might take several hundred years before the compacted salt sealing plug will become impermeable for the flow of brine and the transport of radionuclides.

For the present analysis and the specific case of salt-based repository with boreholes that are sealed with compacted salt plugs the “time of closure” of a borehole plug is a transparent performance indicator for the behaviour of the system. The time of closure may be one of the options in the decision-making process to close a repository.

This indicator is however not a generic parameter since it is only applicable to repository designs that are equipped with compacted salt sealing plugs. Moreover, the indicators “activity flux from compartments” and “radiotoxicity flux from compartment” are indicators that comprise the compaction behaviour of the compacted salt sealing plugs in the present analysis.

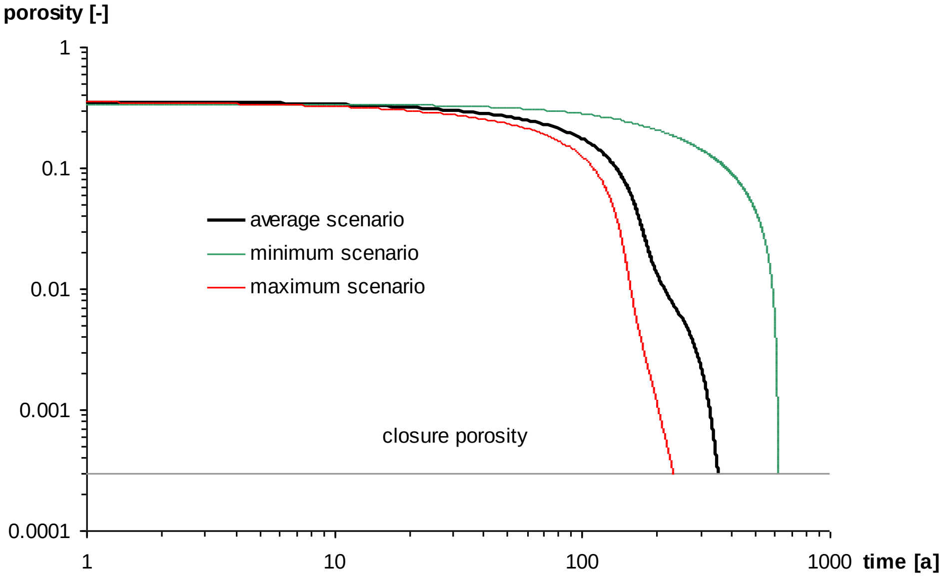

Figure 8.16 Evolution of the porosity of a sealing plug in a repository in salt (NRG).

9. Conclusions

The obtained results show that all proposed safety and performance indicators give useful results. Each indicator has its specific advantages in order to illustrate the properties of a repository system. When used in a complementary fashion, the proposed indicators appear as effective communication tools to present the results of a safety assessment and to explain the functioning of the repository system and the contribution of its safety functions and components. For all considered repository systems the application of multiple safety indicators and performance indicators together provides a much more complete picture of the results of a safety assessment than the effective dose rate alone.

According to the common understanding achieved in this WP the objectives of safety indicators on the one hand and performance indicators on the other hand are very different in a safety assessment. Safety indicators provide statements about the overall safety of a repository system, whereas performance indicators provide information about how the safety is achieved by explaining the functionality of the system. They are very important for the understanding of the role played by different system components.

Because of the fundamental distinction the further conclusions are divided into a section on safety indicators and a section on performance indicators.

9.1 Safety indicators

In PAMINA WP 3.4 the effective dose rate and three complementary safety indicators including their corresponding reference values were identified and tested for all three host rock types (clay, granite, and rock salt) considered within the EU for deep geological repositories. Due to the independently derived reference values the four applied safety indicators provide four different safety statements

- Effective dose rate: Future generations living in the vicinity of the repository will not be exposed at any time to unacceptable concentrations of radionuclides released from the repository. By evaluating carefully different exposure paths and weighting all biological effects to a human individual, the impact on human health by the incorporation of radionuclides released from the repository is found to be insignificant.

- Radiotoxicity concentration in the biosphere water: The safe use of biosphere water as drinking water is not jeopardised by the repository at any time, because the radiotoxicity of biosphere water that can be attributed to the radionuclide flux from the repository is lower than the natural background value found in drinking water.

- Radiotoxicity flux from the geosphere: The radiotoxicity flux from the geosphere to a water body, which represents the interface to the biosphere, is not higher than the present natural radiotoxicity fluxes in this water body.

- Power density in groundwater: The increase of the radioactive power density in the upper aquifer system is not higher than the present natural radioactive power density in groundwater.

The combination of the four indicators and the underlying safety statements gives a strong argument for or against the safety of the repository system. The consistent results found for all four indicators increase the confidence in the outcome of the safety analysis.

The combination of different normalised safety indicators in one figure is a good method to illustrate the outcome of the different safety indicators and the corresponding safety statements. In all considered repository systems the normalised safety indicators are in a very narrow range of about one order of magnitude. This shows that the safety statements given by the different safety indicators are consistent with each other.

Three of the tested indicators were already used in the SPIN project. The conclusions drawn there for granite formations are also valid for clay and salt formations. A new indicator is the power density. It is the only indicator that is not directly related to radiological effects on man, and can be useful to deal with the radiological consequences of the repository on non-human biota. It is a useful complementary safety indicator due to its independence of human properties.

The crucial point of the introduction of a new safety indicator is the determination of an appropriate reference value. In general, the usefulness of such safety indicators should be assessed by the plausibility of the determined reference values.

The derivation of reference values based on natural backgrounds requires a detailed and elaborate analysis of the natural conditions. But these reference values have the advantage to be relatively easy to communicate.

Risk-based indicators can be an important contribution to providing effective and comprehensive indicators for the safety of a repository system because they can be compared with risks of everyday life. However, the assessment of hazards made by individual humans is often very subjective and only partly based on such numerically expressed indicators. Risk should be seen as a useful additional concept for the presentation of the safety of a repository system.

9.2 Performance indicators

Performance indicators are typically inventories, concentrations or fluxes of radionuclides in or between specific parts of the repository system, or other descriptive measures that demonstrate specific properties of the system. Performance indicators are very important for the understanding of the modelled processes and they can be used for the optimisation of the repository system and give valuable arguments for increasing the confidence in the safety of a repository system.

Since radionuclides behave differently, performance indicators based on individual radionuclides can improve the understanding of and the communication about the functionality of the system and its barriers for each radionuclide or decay chain.

For the experts it is essential to transform the massive amount of output data resulting from the simulations of complex repository systems into a limited number of convincing performance indicators to understand how the different barriers act together and where the radionuclides are mainly retained. For communication with licensing authorities as well as with the general public it can be helpful to demonstrate the functioning of the system in an illustrative and understandable way.

The use of time-integrated fluxes is a very illustrative way to present the performance of different compartments. But this type of indicator must be used with care: Because radioactive decay is no longer considered after the integration of the fluxes, the time integral of the flux from a compartment can be some orders of magnitude higher than the activity that really exists outside the compartment. In particular, the effect of short living nuclides may be overestimated here.

Time-integrated radiotoxicity fluxes can also give results for the decay series that are difficult to understand at first sight. The time-integrated radiotoxicity flux leaving a barrier can be greater than the time-integrated radiotoxicity flux entering the barrier. This behaviour is observed in decay chains when an immobile parent radionuclide produces a highly mobile daughter, for instance in the case of the decay of U-238 to Ra-226. To avoid this problem the use of molar activities as weighting scheme can be considered, although this indicator is then less related to human health.

It is recommended always to combine several performance indicators (for instance integrated and non-integrated radiotoxicity fluxes or inventories) to avoid misinterpretations.

Travel times through a compartment can be very useful magnitudes to show the capability of a barrier to delay and limit the releases of a given radionuclide. Travel times should be presented together with the corresponding radionuclide half-lives. Such illustrations provide much information, and clearly identify which radionuclides will decay totally in the barrier and which are expected to cross the barrier. This indicator can also be used at the beginning of a performance assessment to select the radionuclides to be included in the calculations.

To prove and to assess the long-term integrity of a salt barrier, mechanical parameters, especially stress indicators determined in model calculations can be taken into account. In this context two performance indicators are considered: the dilatant state stress and the fluid pressure. For characterisation of both indicators the dilatancy boundary and the fracturing pressure are used. The consideration of these indicators provides information about the integrity of the geological (host rock) barrier of the repository.

In contrast to safety indicators performance indicators are related to specific safety functions and the layout of the repository system. Therefore some of the used performance indicators are very useful for a certain repository system but cannot be transferred to another system. Examples are the combined illustrations of travel times and half-lives (useful for clay and granite but less useful for salt) and indicators for the assessment of the integrity of a salt barrier (not applicable for clay and granite).

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