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Testing of Safety and Performance Indicators (SPIN)

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EXECUTIVE SUMMARY

To assess the overall safety of a geological repository by means of numerical radionuclide release calculations, specific magnitudes are needed which follow from the calculation and can be compared with reference values. Such magnitudes are often called safety indicators. The most common safety indicator used so far is the effective dose rate. The uncertainty of dose rate calculations, however, increases with the time under consideration, and therefore it seems desirable to have additional indicators to improve the reliability of performance assessments. As long as not the overall safety but the performance of individual barriers is assessed, or the functioning of the system is to be showed, another type of indicators can be used, these are sometimes called performance indicators.

The SPIN project identified and tested seven safety and fourteen performance indicators. Safety indicators have been mainly identified by evaluating the open literature, performance indicators through systematic approaches. The indicators have been tested by re-calculating existing performance assessments of disposal systems for high level waste in crystalline formations in Spain, Germany, Finland and Switzerland. The results have been compared and assessed in view of the general applicability of the specific indicators. Although other geological formations than granite have not been tested, the conclusions of the project might be more generally applicable.

The effective dose rate is taken as the basic safety indicator. Two other indicators were found to provide significant benefits and may therefore be used to complement the effective dose rate. The three proposed safety indicators and their preferred application to time frames are:
- *Effective dose rate*: most relevant to early time frames
- *Radiotoxicity concentration in biosphere water*: preference for medium time frames
- *Radiotoxicity flux from geosphere*: preference for late time frames

For the effective dose rate the data from present regulations were used as a range of reference values. Reference values for radiotoxicity concentration and fluxes were taken from nature, based on the assumption that nature in general is radiologically safe. Widely reported concentrations and fluxes corresponding to crystalline sites were used as reference values.

The project concluded that several performance indicators can be used to show different aspects of the functioning of the individual compartments of the multi-barrier system. These indicators and their preferred applications are:
- *Inventories in compartments*: showing where the radionuclides are at different points in time, and the retention of radionuclides from the biosphere
- *Fluxes from compartments*: showing the decreasing release rates from successive compartments, including radioactive decay and ingrowth, and the delayed release
- *Time-integrated fluxes from compartments:* showing decay during delayed transport
- *Concentrations in compartment water*: showing the decrease of concentration by dilution, dispersion and decay in successive compartments
- *Transport times through compartments*: showing the potential importance of individual radionuclides to the release of radiotoxicity by comparing them to their half-lives

For investigations relating to the total radionuclide spectrum, performance indicators based on radiotoxicities should be used. When investigating the behaviour of different types of radionuclides, indicators based on activity are considered appropriate.

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1 INTRODUCTION

High-level radioactive wastes have to be isolated from the biosphere for a very long period of time. For this purpose geological repositories are being considered by many countries, in various host rock formations like salt, clay or granite. Such disposal systems are expected to protect man and the environment from the harmful effects of radionuclides.

A repository which totally isolates its hazardous inventory from the biosphere under any circumstances and for infinite time is obviously safe, but this is a hypothetical case. All management options for long-lived radioactive wastes include some level of risk, and safety has to be assessed by adequate criteria. In general, the relevant safety standards for nuclear activities are established by regulatory agencies or, in some cases by legislation. There is often some variation between the standards applied in different countries, though the general trend is towards harmonised standards, particularly within the European Union.

The safety of a repository has to be established for very long time periods, in order to protect future generations. Long-term performance can not be predicted exactly, because long-term experiments are not possible, and the uncertainty of data about future conditions increases with the length of time. Therefore some 'indications' of the level of safety are needed to enable regulators and society to have confidence in proposed disposal systems. These can be qualitative indications, following, e.g., from an expert's assessment of the geological stability of the formation. Another possibility is to perform numerical calculations. Because the input data are always uncertain to some extent, the results must not be understood as predictions of future development of the repository, but can be a source of quantitative indications of the level of safety. To provide such indications, calculational quantities must be established, such as radiation doses to the most exposed human beings. In this project these calculated quantities are called 'safety indicators'.

A calculated quantity may only provide an indication of the level of safety when it can be compared to a safety-relevant reference value, e.g. a dose constraint. It is a specific and difficult task to find suitable reference values for the different safety indicators. Reference values can be determined, e.g., from natural processes in the environment that is normally accepted to be safe. Reference values can also be given by legal regulations.

The standard safety indicator which has been used for many years in performance assessment studies is the effective dose rate. This is the equivalent dose to an average member of the group of the most exposed individuals in a year. The effective dose rate, however, is calculated using many assumptions about future geosphere and biosphere conditions as well as human behaviour. Such assumptions are uncertain, and the uncertainty increases with the duration of the time period under consideration. By using additional safety indicators the robustness of a safety case may increase, especially if the assumptions on which these indicators depend are less uncertain than those used to determine the dose rate.

Even if safety-relevant reference values are not available, model calculations can nevertheless play an important role in the repository designing process for optimisation of the engineered barrier system. For this purpose some more detailed model output is desirable, in order to analyse how the system works. Such detailed information is not provided by safety indicators because they are designed for an overall assessment of the total system. Therefore, it will be useful to analyse the system with the help of some indicators that allow to quantify the performance of specific subsystems. In this project, such indicators are called performance indicators. The information provided by performance indicators can be used for optimisation of the disposal system, to improve understanding of the role played by different system components and communication of these both to experts and the general public.

Safety and performance indicators have been under discussion for many years. Different indicators have been proposed for use or have been applied in performance assessment studies. Authorities in some countries are considering the introduction of legal regulations that require application of specific safety indicators in repository licensing processes.

The SPIN project was initiated in order to identify suitable safety and performance indicators, and to systematically test them by applying them to recent performance assessment studies and comparing the results. The overall objective of the project was to show the applicability of selected indicators and to point out their specific advantages and disadvantages in practical use, leading to a general assessment of all tested indicators. Four recent national performance assessment studies for granitic host rock have been selected to be re-calculated with regard to the safety and performance indicators to be identified. These studies are:

- ENRESA-2000 (Spain),
- SPA-GRS (Germany),
- Kristallin-I (Switzerland),
- TILA-99 (Finland).

In the first phase of the project, a literature review was done in order to identify indicators which have been proposed or used earlier. Chapter 2 gives a compilation of the proposals. Those indicators which were found to comply with the objectives of the project were selected for further investigation. This initial list was supplemented with indicators identified by the project participants through the application of systematic approaches described in appendices 1 and 2. Ultimately, 7 safety indicators and 14 performance indicators were selected for comparison. The procedure of indicator identification is explained in Chapter 3, the indicators themselves are described in Chapter 4. Chapter 5 gives a compilation of the main characteristics of the four studies.

The calculation of safety indicators does only make sense if reference values are available. A special task of the project was the identification of suitable reference values for the selected indicators. An important source of information was the Co-ordinated Research Programme on safety indicators of IAEA, which is being carried out in parallel to this project. It provides a large amount of geochemical data, measured in different countries and covering very wide ranges of geological conditions. Chapter 6 describes how reference values for SPIN were derived.

Most of the identified indicators were calculated for each study. The results are presented and discussed in Chapters 7 and 8. Finally, an assessment of the indicators was done. The specific advantages and disadvantages of each indicator were identified. Moreover, certain assessment criteria were defined to allow a systematic assessment. For each indicator a conclusion was drawn concerning its possible use in future safety cases. This assessment is contained in Chapter 9 and conclusions are drawn in Chapter 10.

2 INDICATORS FROM LITERATURE REVIEW

To create a basis for the selection of indicators, a literature review has been performed. Many different safety and performance indicators have been considered in the literature. Some authors deal directly with the use of indicators in performance assessments. Such papers typically discuss the specific advantages and disadvantages of indicators with regard to their applicability and meaningfulness, but from a more theoretical point of view. Other publications describe the use of special indicators in performance assessment studies.

Sometimes, a safety indicator is understood as a qualitative or quantitative criterion providing a statement on the safety of a repository, but not directly derived from a performance assessment calculation. Examples for such indicators are groundwater age or the amount of waste in the repository. In other cases, the term 'safety indicator' is more or less identified with a collection of natural concentrations and fluxes in the environment of a repository. For the purpose of the indicator test described here, such values are of interest as reference values, but the indicators themselves must be calculable from the performance assessment output.

Most proposed safety indicators are of the type 'dose', 'risk', 'concentration', or 'flux'. Proposed performance indicators are of different types. Some are fluxes, concentrations or absolute inventories, others are defined in a more specific way, e.g. as a ratio of concentrations in different parts of the repository or at different times. Transport times through barriers are also often proposed as performance indicators.

Most indicators are proposed in identical or very similar forms by several authors. In the literature analysis, it was necessary to identify such equivalences and to classify the proposals with the different comments concerning applicability, advantages and disadvantages. A compilation and classification of the indicators found in the literature is given below. Similar proposals have been aggregated. This chapter only contains information given in the references.

2.1 Safety indicators

The proposals described in this section should be considered as safety indicators, even if this term is not always used by the respective authors.

2.1.1 Indicators of type 'dose rate'

The classical criterion for the assessment of long term safety is dose rate, i.e. the radiological dose to any biota per year. Three different indicators of this type are proposed.

Individual dose rate [Sv/y]

Individual dose rate is the effective dose to a representative human individual in a year. It is calculated in a performance assessment by the consideration of relevant exposure pathways. Assumptions about the biosphere, including locations and lifestyles of future generations, are necessary, and dose coefficients are needed. The indicator is widely used and allows an assessment of individual health risk.

Some authors consider individual dose rate not adequate for human intrusion scenarios, because the dose to single individuals can widely exceed the dose constraint.

References: [1], [2], [3], [4], [5], [6], [7], [8]

Collective dose rate [person-Sv/y]

Collective dose rate is the effective dose to the presumed regional population in a year. It is calculated in performance assessment by the consideration of relevant exposure pathways. Calculation requires the same data and assumptions as individual dose rate, and additionally predictions about future population distribution.

References: [1], [4], [8]

Dose rate to animals and plants [Gy/y]

Dose rate is calculated with special 'lifestyle' assumptions for a few selected organisms. Four generic target organisms have been proposed: a plant, a mammal, a bird and a fish. Calculating the potential radiological effects on non-human biota will help to demonstrate that the effect of a repository on the environment, not only on man, is acceptable. It can only be used as an additional indicator, since the main protection goal is human health.

References: [9]

2.1.2 Indicators of type 'risk'

Indicators of the 'risk' type take into account the probabilities of all significant scenarios and the probability of damage to individuals. It is above all useful to calculate risks when performing probabilistic assessment. Such indicators have not been selected for further assessment, but they are described here for completeness.

Individual risk [1/y]

Individual risk is the probability of death or serious health or genetic defect, caused by effects of radionuclide release from the repository, for a human individual or his/her descendants in a year. It is calculated under consideration of exposure probability for a typical member of a 'critical group', and summed over all significant scenarios. It can be compared with risks from other sources, like smoking, car-driving or using airplanes. All scenarios and exposure pathways are taken into account. It is an abstract and uncertain quantity which cannot be measured and may be difficult to communicate.

References: [10], [11], [2], [3], [4], [12], [5], [6], [13], [8]

Societal risk [1/y]

Societal risk is the probability of exceeding some number of fatalities, e.g. 10 or more people die, in a given year, due to effects of radionuclide release from the repository. Calculation requires the same data and assumptions as individual risk, and additionally predictions about future population distribution.

References: [10], [12], [8]

2.1.3 Indicators of type 'concentration'

Concentrations can be calculated for different locations in the repository system or the biosphere, and compared with measured natural data. These indicators are independent of uncertain assumptions on future generations, but can be sensitive to variations in groundwater or surface water flow.

Concentrations are normally determined for single nuclides. For a safety indicator, however, it is necessary to sum over all nuclides to get an integral measure, directly related to safety. Since it is not very meaningful to add the activities or molar concentrations of different nuclides, they must be weighted by suitable coefficients, such as the ingestion dose coefficients (IDC) provided by ICRP [14]. The sum of nuclide activities, each multiplied by the relevant ingestion dose coefficient, is called radiotoxicity and measured in Sievert (Sv). Therefore, all proposed safety indicators of concentration type are presented here in the form of radiotoxicity concentrations. All concentration indicators have common advantages and disadvantages. They are conceptually simple indicators which are easy to understand and communicate. A biosphere model is not needed for calculation.

Concentration in groundwater [Sv/m^3]

The radiotoxicity concentration in near-surface groundwater is calculated from a near field model and a far field model. It is widely independent of assumptions about the interface to biosphere.

References: [9], [15], [2], [6], [16]

Concentration in biosphere water [Sv/m^3]

The biosphere water is the water used by man for drinking, cattle watering, irrigation, etc. It is normally taken from a well or a river. The concentration in this water is more directly related to human health than the concentration in groundwater. For calculation, specific assumptions about the interface to the biosphere are necessary.

References: [9], [2], [6], [16]

Concentration in soil [Sv/kg]

The concentration in soil gives an indication of the quantity of nuclides accumulated in solid soil over a long period. For calculation, some chemical effects like sorption and precipitation have to be considered. This indicator considers radiotoxicity accumulation over long times but it is not directly related to human health.

References: [9], [2], [6], [16]

Concentration in air [Sv/m^3]

Concentration in air is an indicator for the effects of re-suspended soil, gaseous radionuclides and aerosols. Instead of the ingestion dose coefficients, the inhalation dose coefficients must be used for calculations. These coefficients are also provided by ICRP. This is the only indicator which considers gaseous radionuclides, but it is incomplete. It can be used only in connection with other indicators.

References: [9], [6], [16]

2.1.4 Indicators of type 'flux'

Many proposed indicators are contaminant fluxes. Since a safety indicator must provide a safety-related measure for the entire system, only the radiotoxicity flux (or release) from the repository to the biosphere can be considered a safety indicator. The basic reasoning for using radiotoxicity (instead of, e.g., activity) is the same as given in the previous section for concentrations. Natural fluxes of radionuclides are a good basis to derive reference values for this indicator.

Radiotoxicity release [Sv/y]

This is the flux of radiotoxicity from the repository to the biosphere. It is calculated from nuclide-specific activity fluxes by multiplying with the ingestion dose coefficients and summing. It is a robust indicator which is independent of uncertain data and assumptions about biosphere, but not directly related to human health. Factors like population density and water supply are not considered.

References: [1], [2]

2.2 Performance indicators

The indicators presented in this section are considered as performance indicators, although this terminology is not always used by the authors of the considered papers. In this project performance indicators are distinguished from safety indicators as discussed in Chapter 3.

The proposals in the literature not always discriminate between safety and performance indicators. Although calculated doses are only relevant to the performance of the repository system as a whole, some indicators can be determined for repository sub-systems. For example, radionuclide concentrations can be calculated for each compartment, and radionuclide fluxes can be calculated at the boundary between compartments. Both indicators can be calculated for single nuclides, and therefore, also activity or molar concentrations or fluxes can be considered.

2.2.1 Indicators of type 'flux'

Fluxes inside the repository may provide interesting information about the effectiveness of the barrier system, especially if they are compared at different locations. All proposed indicators of this type are either activity fluxes or relative fluxes normalised to the initial inventory of the relevant nuclide. Therefore, these proposals can be treated together and presented as one performance indicator.

(Relative) activity flux through compartment boundaries [Bq/y or 1/y]

The activity flux of a specific nuclide from a compartment is the amount of activity that comes out of that compartment in a year, i.e. the flux from the relevant compartment to the next one. These fluxes are normally calculated within a performance assessment in the nuclide transport calculations. The relative flux is obtained by dividing by the initial inventory and is easy to calculate. Comparison of fluxes allows a statement on the effectiveness of the barrier system, such calculations are useful in site investigation and design.

References: [4], [2], [17], [6], [8]

2.2.2 Indicators of other types

Several performance indicators have been proposed which cannot be attributed to any of the types considered so far. Some authors propose indicators of the type 'time', but it is not always clear what is meant exactly. Other indicators are somewhat more sophisticated, to yield very specific information. Some of these proposals are either site-specific, or often not suitable for a general analysis for other reasons. Such proposals are omitted here. Other proposals which are similar to one another, have been grouped together for presentation in this section.

Nuclide transport time [y]

The transport time is the time a nuclide needs to pass through a compartment. It is not easy to calculate, because a concentration front is normally spread during the transport, and it is difficult to define unique points of time for entering and leaving the compartment, especially if radioactive decay plays a role during the transport time. It is a clear measure which gives a direct impression of the retardation effect of the compartment. Transport times can be compared with the nuclide's half-life. This gives a rough estimation of whether or not the nuclide will largely decay within the compartment.

References: [2], [4], [6], [16]

Radionuclide distribution in the disposal system [-]

For a compartment, this indicator consists of three dimensionless values: the inventory present in the compartment, the inventory decayed in the compartment, and the inventory released from the compartment after some time, each divided by the nuclide's initial inventory. They can be calculated from intermediate results of the performance assessment calculation. For a specified point of time, this indicator clearly illustrates the distribution of the nuclide with regard to a specific compartment. Figures can be useful for explaining results to non-technical audiences.

References: [9], [15], [18]

Compartment / barrier effectiveness [-]

A unit amount of a radionuclide is assumed to enter the compartment at time $t = 0$. The fraction that remains in the compartment after some fixed period of time, e.g. 10^4 or 10^5 years, is calculated. This is the dimensionless compartment effectiveness for the nuclide. It can be calculated from intermediate results of a performance assessment calculation. The value is 1 or smaller. It gives an impression of the retention capability of the compartment. The product of effectiveness of consecutive compartments/barriers can be used as a conservative estimator for the effectiveness of the barrier system.

References: [9]

3 IDENTIFICATION AND SELECTION OF INDICATORS

During the review of the open literature the indicators which have been considered by other authors were identified. Additional indicators were identified by the participants of the project based on individual conceptions and also by the use of systematic approaches described in Appendices 1 and 2. The testing of indicators in this project is limited to those that can be calculated using performance assessment models. Hence, a selection was made for the purpose of this project.

3.1 Definition of Indicators

A definition for indicators is given by the International Atomic Energy Agency [4], [19] which also introduces different classes of indicators. From this definition, which is accepted as a general basis also for this project, only the terms 'safety indicator' and 'performance indicator' are used. Their understanding in this report is as follows.

A safety indicator must
- provide a measure of the safety of the whole system,
- allow a comparison with a safety-relevant reference value,
- take into account the contributions of all radionuclides,
- be calculable using performance assessment models.

A performance indicator must
- provide a measure of the performance of the whole system or a subsystem,
- allow a comparison between different options or with technical criteria,
- take into account the contributions of all radionuclides or a single radionuclide,
- be calculable using performance assessment models.

The indicators can be time-dependent or constant. When comparing a time-dependent safety indicator to a reference value, the maximum value of the indicator is often used. Subsystems considered for performance indicators consist of one or more of the barriers or of parts of a barrier.

The given requirements have to be met by all indicators selected for use in the project.

3.2 Selection of Safety Indicators

Safety indicators must provide a measure on the safety of the whole disposal system. In this project the effective dose rate was selected as the basic indicator, as it has been used in most safety assessments up to now. For calculations representing long time frames the dose rate is an indicator of the expected level of safety of the system and is not intended to give a prediction of future consequences. Reference values for dose rates are commonly available and often given by national regulations.

For the dose rate the biosphere dose conversion factors are applied as weighting for the integration over all radionuclides. Other indicators need a different weighting scheme. A possible set of weighting factors is given by the ingestion dose coefficients developed by ICRP [14], which do not include assumptions about biosphere pathways. Applying ingestion dose coefficients, concentrations, fluxes and total amounts are derived in terms of radiotoxicities. Reference values for indicators based on radiotoxicity can be provided in general.

Other weighting schemes are based on radionuclide-specific data used as reference values directly. This can be applied again on the basis of concentrations, fluxes and total amounts. No further integrated reference values are necessary in this case. The development of weighting schemes including all radionuclides is under consideration, but it will be difficult for radionuclides not occurring in nature.

Seven safety indicators as given in TABLE 3-1 have been selected. For the total radiotoxicity two indicators have been introduced based on two potential calculation schemes. One is the radiotoxicity outside the geosphere that takes into account the decay and ingrowth of radionuclides released from the geosphere. The other is the time-integrated radiotoxicity flux from the geosphere, not considering the decay and ingrowth of radionuclides after their release from the geosphere.

TABLE 3-1 Safety Indicators Selected

Safety Indicator	Unit
Effective dose rate	Sv/y
Radiotoxicity concentration in biosphere water	Sv/m^3
Radiotoxicity flux from geosphere	Sv/y
Time-integrated radiotoxicity flux from geosphere	Sv
Radiotoxicity outside geosphere	Sv
Relative activity concentration in biosphere water	-
Relative activity flux from biosphere	-

3.3 Selection of Performance Indicators

The performance of a repository system is provided by the individual barriers of the multi-barrier system. Performance indicators therefore can be derived through the analysis of quantities describing the behaviour of the radionuclides in the individual barriers. In this project such an approach is identified as A0. These quantities are

- the total amount of hazardous material in the barrier,
- the flux of hazardous material from the barrier,
- the released amount of hazardous material from the barrier,
- the concentration of hazardous material in the barrier,
- the transport time of the hazardous material through the barrier.

The variations of these quantities from barrier to barrier show the performance of the multi-barrier system. They are therefore considered as performance indicators. The interpretation of the indicators has to take into account the time dependence of most of the quantities. Simpler representations can be derived by considering maximum values in each of the barriers.

Performance indicators can also be derived through a systematic approach using safety functions. A safety function is a specific feature of the barriers or a specific process in the barrier system which contributes to the safety of the repository system. In the definition of safety functions, the following phenomena have been considered:

- isolation or physical containment for the whole time under consideration,
- delay of radionuclide transport and radioactive decay during that time, due to
  - isolation or physical containment for a limited period of time,
  - slow release of radionuclides from the waste form,
  - retardation of radionuclides in precipitates,
  - slow transport of radionuclides by diffusion only or by slow advection,
  - retardation by sorption onto the buffer material or on the rock,
- dispersion and dilution of radionuclides in the near field, the far field and the biosphere.

To identify performance indicators by using safety functions two approaches have been applied. The first approach identified a large number of indicators illustrating the functioning of the repository, allowing for some overlap between the indicators concerning the safety functions. In the project this approach is identified as A1 and is presented in Appendix 1. The other approach is aimed at identifying a minimum number of indicators exactly representing the three safety functions 'isolation', 'delay and decay', and 'dispersion and dilution'. In the project this approach is identified as A2 and is presented in Appendix 2.

The performance indicators selected on the basis of the three approaches are given in TABLE 3-2 also showing which approach identified which indicator. The performance indicators are related to compartments of the disposal system. The compartments considered are introduced in Section 3.4.

The approach A0 is based on five different features of the disposal system. Due to multiple options concerning the definition of the hazardous material eleven performance indicators have been selected. For the amount of radionuclides in the compartment, the flux from the compartment, the released amount from the compartment and the concentration in the compartment, both the activity and the radiotoxicity are considered. For the quantity of hazardous material outside the compartment again two indicators have been introduced accounting for two potential calculation schemes. One calculation scheme is the amount of radionuclides outside the compartment, that takes into account radioactive decay/ingrowth after leaving the compartment and the other calculation scheme is the time-integrated flux from the compartment, that doesn't.

The transport times through compartments are investigated for single radionuclides.

TABLE 3-2 Performance Indicators Selected

Performance Indicator	Unit	A0	A1	A2
Activity in compartments	Bq	x	x	
Activity flux from compartments	Bq/y	x	x	
Time-integrated activity flux from compartments	Bq	x	x	
Activity outside compartments	Bq	x	x	
Radiotoxicity in compartments	Sv	x	x	
Radiotoxicity flux from compartments	Sv/y	x	x	
Time-integrated radiotoxicity flux from compartments	Sv	x	x	
Radiotoxicity outside compartments	Sv	x	x	
Activity concentration in compartment water	Bq/m^3	x	x	
Radiotoxicity concentration in compartment water	Sv/m^3	x	x	
Transport time through compartments	y	x	x	
Proportion of not totally isolated waste	-			x
Time-integrated flux from geosphere / initial inventory	-			x
Concentration in biosphere water / waste package water	-			x

The activity outside the compartments and the radiotoxicity concentrations in compartment water have not been considered during the calculational exercises. The activity outside compartments provides little additional information compared to the activity in compartments, and the consideration of radiotoxicity outside compartments in addition to radiotoxicity in compartments was felt sufficient. The activity concentrations were calculated only for a few selected compartments.

The A1-approach in general identified the same indicators as the A0-approach and thus gave some backup to the selection. However, not all of the indicators identified in this approach have been further investigated, some are considered as parameters of the repository system. The A2-approach identified three additional indicators, although the *proportion of not totally isolated waste* is a design parameter and *time-integrated flux divided by initial inventory* provides similar information as the *time-integrated flux from compartments* itself.

3.4 <u>Selection of Compartments</u>

The definition of performance indicators is related to compartments and not to barriers in order to allow for a higher flexibility and not to use the term 'barrier' for subsystems which are not typical barriers. Altogether seven compartments were selected, though it was not considered useful to calculate each performance indicators for each compartment. The compartments and their application to different types of performance indicators are given in TABLE 3-3.

TABLE 3-3 Selection of Compartments for Types of Performance Indicators

Compartments	Amount in	Amount outside Flux from Time-integrated flux from	Concentration in Water	Transport Time through
Waste form	x	x		
Precipitate	x			
Waste package		x	x	
Buffer	x			x
Near field		x		
Geosphere	x	x		x
Biosphere	x		x	

The compartments according to TABLE 3-3 have been defined for a repository in hard rock formations. The barrier system for such a repository consists of the waste form, the container, the buffer and the geosphere. The container together with its content is called waste package. The waste package water refers to all the liquid in the waste package that contains radionuclides in a soluble form. The precipitate includes all the radionuclides in the waste package which are neither in the waste form, nor dissolved in the water. The near field comprises the buffer and the waste package.

4 DESCRIPTION OF SELECTED INDICATORS

In total, seven safety and fourteen performance indicators were identified. For defining the indicators the following nomenclature is used:

- **Radionuclides** are numbered by n.

- The **ingestion dose coefficient** D_n is the dose caused by ingestion of radionuclide n (Sv per ingested Bq). The ingestion dose coefficients of ICRP 72 [14] for adults, that correspond to the committed effective dose integrated over 50 years, are used. The effects of daughters produced in vivo are accounted for in the ICRP 72 ingestion dose coefficients.

- **Biosphere water** denotes the water which is used by man for drinking, feeding livestock or irrigation and which receives the releases of radionuclides from the geosphere. It is assumed to be taken at a given point which depends on the characteristics of the study and can be a well, a river, a lake, or something similar.

- The **biosphere dose conversion factor** B_n is the annual dose to the most exposed members of the public (so-called critical group) caused by an unit concentration of radionuclide n in the biosphere water. It is measured in [(Sv/y)/(Bq/m^3)]. It takes into account different exposure pathways as well as living and nutrition habits. Biosphere dose conversion factors are provided by the biosphere analyses, following the guidance given in national regulations where available. The biosphere dose conversion factors used are presented in TABLE 5-6.

- c_n is the **activity concentration** [Bq/m^3] of radionuclide n in the biosphere water.

- s_n is the **activity flux** [Bq/y] of radionuclide n from the geosphere to the biosphere.

- $a_{n,i}$ is the **activity** [Bq] of radionuclide n in compartment i.

- $c_{n,i(\text{water})}$ is the **activity concentration** [Bq/m^3] of radionuclide n in the water of compartment i.

- $s_{n,i}$ is the **activity flux** (Bq/y) of radionuclide n from compartment i.

- $a_{n,in}$ is the **initial activity inventory** of radionuclide n that has the potential to be released.

All indicators deal with radionuclides disposed of in the repository, including their daughters. Radionuclides naturally present in the disposal system are not considered. The contributions of short-lived daughters are added to those of their long-lived precursors, assuming secular equilibrium. For most indicators, a total value summed over all radionuclides as well as the values of the most important nuclides are presented. Some of the indicator values may also be separately summed over nuclides emitting α-radiation and nuclides emitting β- or γ-radiation.

4.1 Safety indicators

All safety indicators described below are time-dependent quantities which typically are presented vs. time, together with their reference values.

Effective dose rate [Sv/y]

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 enrichment in the biosphere, different exposure pathways as well as living and nutrition habits.

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

Radiotoxicity concentration in biosphere water [Sv/m³]

The indicator represents the radiotoxicity of the radionuclides in 1 m^3 of biosphere water. It also can be understood as the dose which is received by drinking of 1 m^3 of biosphere water.

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

Radiotoxicity flux from geosphere [Sv/y]

The indicator represents the radiotoxicity of the radionuclides released from the geosphere to the biosphere in a year. It can also be understood as the annual dose to a single human being who would ingest all radionuclides released from the geosphere to the biosphere.

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

Time-integrated radiotoxicity flux from geosphere [Sv]

The indicator presents the cumulated radiotoxicity flux from the geosphere to the biosphere. It can also be understood as the cumulated radiological impact due to continuous ingestion of all radionuclides released from the geosphere to the biosphere.

Calculation: $$\int_0^t \sum_{\text{all nuclides } n} s_n(\tau)\, D_n \, d\tau$$

Radiotoxicity outside geosphere [Sv]

The indicator represents the total radiotoxicity in the biosphere due to the releases from the repository without taking into account any removal processes in biosphere, such as sedimentation. The difference to the previous indicator is that radioactive decay and ingrowth in the biosphere are taken into account.

Calculation: $$\int_0^t \sum_{\text{all nuclides } n} s_n(\tau)\, D_n \, d\tau - \text{decay} + \text{ingrowth}$$

Relative activity concentration in biosphere water [-]

The indicator compares the concentrations of radionuclides in the biosphere water with reference values, such as concentrations of natural radionuclides or constraints specified by the regulator. The weighted sum of concentrations should be less than 1.

Calculation: $$\sum_{\text{all nuclides n}} \frac{c_n}{c_{n,ref}}$$

Relative activity flux from geosphere [-]

The indicator compares the activity flux from the geosphere to the biosphere with reference values, such as natural fluxes of radionuclides or constraints specified by the regulator. The weighted sum of fluxes should be less than 1.

Calculation: $$\sum_{\text{all nuclides n}} \frac{s_n}{s_{n,ref}}$$

4.2 Performance indicators

The performance indicators described below are mostly time-dependent quantities. Sums can be evaluated over all radionuclides or a selected group of nuclides, e.g. α- or β/γ- emitters.

Activity in compartments [Bq]

The indicator represents nuclide-specific activities as well as the total activities summed over α-emitters, β/γ-emitters or all radionuclides in the compartments. The considered compartments are waste form, precipitate, buffer, geosphere, and biosphere.

Calculation: $$a_{n,i} \;;\quad \sum_{\text{nuclides } n} a_{n,i} \;,$$

Activity flux from compartments [Bq/y]

The indicator represents the activity flux from a compartment for single radionuclides as well as summed over α-emitters, β/γ-emitters or all radionuclides. Activity fluxes from waste form, waste package, near field, and geosphere, are calculated.

Calculation: $$s_{n,i} \;;\quad \sum_{\text{nuclides } n} s_{n,i}$$

Time-integrated activity flux from compartments [Bq]

The indicator represents the cumulated activity flux from compartment *i* to *i*+1 for single radionuclides as well as summed over α-emitters, β/γ-emitters or all nuclides. Time-integrated activity fluxes from waste form, waste package, near-field, and geosphere are calculated.

Calculation: $\int_0^t s_{n,i}(\tau)\,d\tau$; $\sum_{\text{nuclides } n}\left(\int_0^t s_{n,i}(\tau)\,d\tau\right)$

Activity outside compartments [Bq]

The indicator represents the activities outside the compartments for single radionuclides as well as summed over α-emitters, β/γ-emitters or all radionuclides. Activities outside the waste form, waste package, near field, and geosphere are calculated.

Calculation: $\sum_{j>i} a_{n,j}$; $\sum_{\text{nuclides } n}\left(\sum_{j>i} a_{n,j}\right)$

Radiotoxicity in compartments [Sv]

The indicator represents nuclide-specific radiotoxicities as well as the total radiotoxicities in the compartments. The considered compartments are waste form, precipitate, buffer, geosphere, and biosphere. The radiotoxicity in the biosphere is equal to the safety indicator 'radiotoxicity outside geosphere'.

Calculation: $a_{n,i} D_n$ and $\sum_{\text{nuclides } n} a_{n,i} D_n$

Radiotoxicity flux from compartments [Sv/y]

The indicator represents the radiotoxicity flux from compartment *i* for single radionuclides as well as summed over all radionuclides. Radiotoxicity fluxes from the waste form, waste package, near field, and geosphere are calculated.

Calculation: $s_{n,i} D_n$ and $\sum_{\text{nuclides } n} s_{n,i} D_n$

Time-integrated radiotoxicity flux from compartments [Sv]

The indicator represents the cumulated radiotoxicity flux from a compartment for single radionuclides as well as summed over all radionuclides. Time-integrated radiotoxicity fluxes from the waste form, waste package, near field, and geosphere are calculated. The time-integrated radiotoxicity flux from the geosphere is also a safety indicator.

Calculation: $\int_0^t D_n s_{n,i}(\tau)\,d\tau$ and $\sum_{\text{nuclides } n}\left(\int_0^t D_n s_{n,i}(\tau)\,d\tau\right)$

Radiotoxicity outside compartments [Sv]

The indicator represents nuclide-specific radiotoxicities as well as the total radiotoxicities outside the compartment considered. Radiotoxicities outside the waste form, waste package, buffer, and geosphere are calculated.

Calculation: $\sum_{j>i} a_{n,j} D_n$ and $\sum_{\text{nuclides } n} \left(\sum_{j>i} a_{n,j} D_n \right)$

Activity concentration in compartment water [Bq/m³]

The indicator represents nuclide-specific activity concentrations as well as the total activity concentration in the water of the compartments.

Calculation: $c_{n,i(\text{water})}$; $\sum_{\text{nuclides } n} c_{n,i(\text{water})}$

The concentrations have been calculated for the water inside the defective waste package and for the biosphere water.

Radiotoxicity concentration in compartment water [Sv/m³]

The indicator represents nuclide-specific radiotoxicity concentrations as well as the total radiotoxicity concentrations in the water of the compartments.

Calculation: $c_{n,i(\text{water})}$; $\sum_{\text{nuclides } n} c_{n,i(\text{water})}$

These concentrations have not been calculated in the project .

Transport time through compartments [y]

Transport times through the buffer and the geosphere are calculated for single nuclides without taking radioactive decay into account. The comparison of the calculated transport times and half-lives of the radionuclides facilitates an evaluation of the effectiveness of these barriers in delaying radionuclide transport. Results are presented in a diagram showing half-lives versus transport times.

Calculation: Concentrations of nuclides are kept constant at the inner boundaries of the buffer and the geosphere. This will finally result in a steady-state release rate from the barrier. The transport time is defined as the time after which half of the steady-state release rate is reached.

Proportion of not totally isolated waste [-]

This indicator corresponds to the safety function 'physical confinement' in the sense of total isolation for the entire period under consideration. It is the fraction of waste inventory which can contribute to the release of contaminants, e.g. by failure of only one container out of thousand. The indicator is represented by a single value.

Calculation: Not needed, design parameter.

Time-integrated flux from compartments divided by initial inventory [-]

This indicator corresponds to the safety function 'delay and decay'. It shows the fraction of initial inventory, i.e. the inventory after repository closure, released after a given time, either in terms of activity of single nuclides or total radiotoxicity. Its value after infinite time shows the total fraction of a nuclide's inventory or the total radiotoxicity that is released. The complementary of that fraction has been decayed during the time of delay.

Calculation: $\frac{1}{a_{n,\mathrm{in}}} \int_0^t s_{n,i}(\tau)\, d\tau \quad ; \quad \sum_{\text{nuclides } n} \left(\int_0^t s_{n,i}(\tau)\, D_n\, d\tau \right) / \sum_{\text{nuclides } n} a_{n,\mathrm{in}} D_n$

Concentration in Biosphere water divided by concentration in waste package water [-]

This indicator corresponds to the safety function 'dispersion and dilution'. It shows the ratio between mass concentrations in biosphere water and waste package water. The indicator is intended to show the integrated dilution effect of the near field, the geosphere and the biosphere, uncoupled from decrease in concentration due to decay and a finite release time from the water package. Therefore, it is calculated for stable isotopes. If calculated with a constant concentration in the waste package water, it reaches an asymptotic value which shows the total dilution by the barriers outside the waste package.

Calculation: $c_{n,\mathrm{Bios}} / c_{n,\mathrm{W}P}$

5 DESCRIPTION OF PERFORMANCE ASSESSMENT STUDIES

For the testing of indicators, four performance assessment studies of repositories in granite formations have been considered. Technical reports are available for these studies, as listed below. In this chapter, some common characteristics of the repositories are described, followed by a specific description of the individual studies.

TABLE 5-1 Overview of studies participating in the SPIN project

Organisation	Study
GRS	SPA-GRS: Spent fuel performance assessment (SPA) for a hypothetical repository in crystalline formations in Germany. [20]
ENRESA	ENRESA 2000. Safety and Performance Assessment of a Spent Fuel Repository in a Granitic Formation. December 2001. [21]
Nagra/Colenco	Kristallin-I Safety Assessment Report. [22]
VTT	TILA-99: Safety assessment of spent fuel disposal in Hästholmen, Kivetty, Olkiluoto and Romuvaara. [23]

5.1 General description of studies

TABLE 5-2 gives an overview of general data. Three of the studies consider direct disposal of spent fuel elements either in vertical boreholes or in horizontal drifts. One study considers disposal of vitrified high-level waste from the reprocessing of spent fuel. In this case, the waste inventory is specified as the equivalent mass of heavy metal being reprocessed.

TABLE 5-2 General data of the studies used for SPIN

Parameter	SPA-GRS	ENRESA-2000	Kristallin-I	TILA-99
Waste type	spent fuel	spent fuel	vitrified HLW	spent fuel
Burn-up [GWd/t_{HM}]	45	40	33	36
Total waste inventory [t_{HM}]	25 000	6 640	equiv. 3 730	2 600 - 4 000
Content of a container [t_{HM}]	1.6	1.845	equiv. 1.4	2.14
Type of container	steel	carbon steel	steel	copper-iron
Container lifetime [y]	1 000	1 300	1 000	0 or 10 000*
Emplacement technique	boreholes	drifts	drifts	boreholes
Depth of repository [m]	900	500	1000	500

(*)depending on scenario (SH or DC)

The studies assume different fractions of the total waste inventory contributing to the release. While ENRESA-2000 and Kristallin-I both use the entire inventories, SPA-GRS assumes that 25% of the inventory participates in the release via fast transport paths, the rest being retained in slow transport paths. TILA-99 calculates the release from only one container.

TABLE 5-6 lists all nuclide-specific data, including the inventories of all nuclides for the time of repository closure. TABLE 5-7 lists all element-specific data, such as Kd-values and solubilities. The data of these tables will be referred to later in this chapter.

5.1.1 General characteristics of the near field

The near field comprises the waste package, the surrounding buffer material and, in some studies, the excavation disturbed zone (EDZ). All emplacement drifts and boreholes are of cylindrical shape with the cylindrical containers in the centre. A cross-section of the near field is schematically shown in FIGURE 5-1 for an emplacement in vertical boreholes.

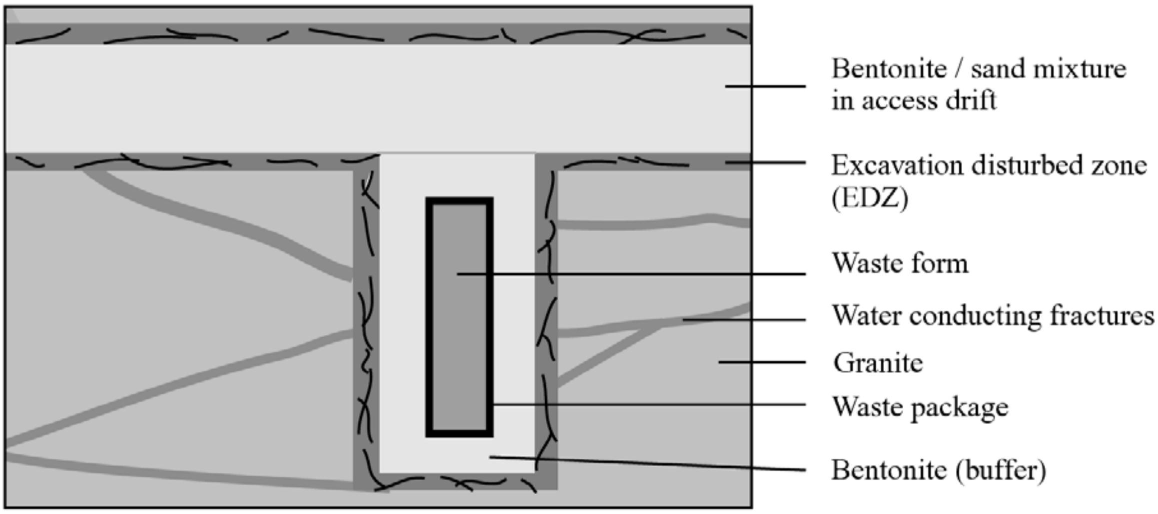

FIGURE 5-1 Schematic view of the near field for borehole emplacement

A borehole contains a single waste package, while each emplacement drift contains several waste packages. Each package is entirely surrounded by a bentonite buffer. Access drifts are backfilled with a mixture of bentonite and sand. Access of water to the emplacement borehole or drift is via the water conducting features in the host rock and the excavation-disturbed zone (EDZ). Transport of radionuclides after release from the waste packages is by diffusion through the bentonite buffer and by advective transport through the EDZ and the water conducting features. The main characteristics of the near field of the different studies are summarised in TABLE 5-3. Details of the data for the studies under consideration are given in Sections 5.2 - 5.5.

TABLE 5-3 Selected near field data (n.a. = not applicable)

Parameter	SPA-GRS	ENRESA	Kristallin-I	TILA-99
Total number of containers	15 605	3 600	2 693	1400 - 2500
Number of containers used in calculation	3 900	3 600	2 693	1
Container diameter [m]	0.53	* 0.90	0.94	1.05
Borehole or drift diameter [m]	1.2	2.4	2.5	1.75
Radial thickness of bentonite buffer [m]	0.33	0.725	0.73	0.35
Total porosity of bentonite	0.38	0.407	0.38	0.43
Near field flow [litres/container and year]	1.0	0.06	1.1	cf. sec 5.5
Darcy velocity in EDZ [m/y]	n. a.	$2.3\cdot10^{-5}$	n. a.	cf. sec 5.5
Darcy velocity in granite [m/y]	$3.5\cdot10^{-5}$	$2.3\cdot10^{-6}$	$2.6\cdot10^{-5}$	cf. sec 5.5
Pore diffusion coeff. in bentonite [m^2/y]	$1.58\cdot10^{-2}$	$3.15\cdot10^{-3}$	$1.7\cdot10^{-2}$	** $3.15\cdot10^{-3}$

(*) Containers are emplaced inside a carbon steel liner of 0.95 m outer diameter. (**) for neutral species

5.1.2 General characteristics of the far field

The far field is the geosphere outside the EDZ up to the biosphere. FIGURE 5-2 gives a schematic view of the far field in granite. The emplacement drifts and boreholes, denoted as repository, are assumed to be situated in an area of the granite body without major water conducting faults. Thus, the far field transport is modelled between the repository and such a major water conducting fault, assuming fracture flow through small water conducting features with matrix diffusion into the wall rock. ENRESA-2000 far field transport model includes all the formation from the repository to the biosphere, leading to a longer transport path. The main parameters for these transport calculations are listed in TABLE 5-4.

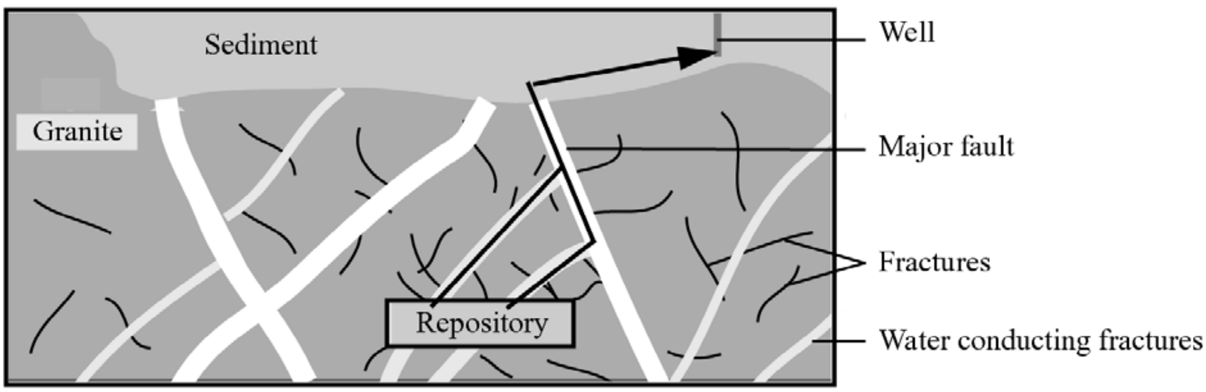

FIGURE 5-2 Schematic view of the far field

TABLE 5-4 Selected far field data (n.a. = not applicable)

Parameter	SPA-GRS	ENRESA	Kristallin-I	TILA-99
Length of transport path [m]	200	4 600	200	600
Darcy velocity [m/y]	$3.5 \cdot 10^{-5}$	$5.5 \cdot 10^{-5}$	$2.6 \cdot 10^{-5}$	$1.2 \cdot 10^{-3}$
Water travel time [y]	50	8 400	100	25
Peclet number [-]	10	10	10	Infinite
Kinematic porosity [-]	$8 \cdot 10^{-6}$	$1 \cdot 10^{-4}$	$1.4 \cdot 10^{-5}$	$5 \cdot 10^{-5}$
Discharge flow from the far field [m^3/y]	140	n. a.	3	n. a.
Parameters of fractures and porous medium in the altered wall rock				
Width of flow channels per area [m/m^2]	0.01	0.05	0.014	0.1
Aperture [m]	$8.0 \cdot 10^{-4}$	$2.0 \cdot 10^{-3}$	$1.0 \cdot 10^{-3}$	$5.0 \cdot 10^{-4}$
Porosity of altered wall rock	0.005	0.005	0.01	0.001 - 0.005
Penetration depth [m]	0.02	0.05	0.02	0.1
Pore diffusion coefficient [m^2/y]	$1.0 \cdot 10^{-3}$	$3.15 \cdot 10^{-4}$	$1.0 \cdot 10^{-3}$	$1.6 \cdot 10^{-4}$ - $7.9 \cdot 10^{-4}$

5.1.3 General characteristics of the biosphere

In the studies SPA-GRS, ENRESA-2000 and Kristallin-I a detailed biosphere analysis has been performed taking into account exposition paths via drinking water, various ingestion pathways (incl. irrigation), and external radiation. In TILA-99 only the drinking water path has been taken into account. In SPA-GRS and Kristallin-I, biosphere dose conversion factors have been derived from the biosphere transport calculations using national data and previous ICRP dose factors [24], [25]. ENRESA-2000 also applies biosphere dose conversion factors, but has calculated them from new ICRP ingestion factors and inhalation factors [14].

The biosphere dose conversion factors are compiled in TABLE 5-6. Applying these factors, a radionuclide concentration in biosphere water can be converted to a dose. The ingestion dose coefficients used for calculating radiotoxicities are also listed in TABLE 5-6, column 'IDC-SPIN'. They are based on new ICRP values, as listed in column 'ICRP 72' [14]. The dilution factors listed in TABLE 5-5 were applied to calculate concentrations in the biosphere water from concentrations at the end of near field.

TABLE 5-5 Parameter of the biosphere (n.a. = not applicable)

Parameter	SPA-GRS	ENRESA	Kristallin-I	TILA-99
Volumetric flow in the biosphere [m^3/y]	$8.0 \cdot 10^{+6}$	$1.0 \cdot 10^{+6}$	$5.5 \cdot 10^{+6}$	$1.0 \cdot 10^{+5}$
Dilution factor from near field to biosphere	$2.1 \cdot 10^{+6}$	$5.0 \cdot 10^{+6}$	$1.8 \cdot 10^{+6}$	n. a.

5.2 SPA-GRS

The GRS study has been performed within the framework of the SPA project of the European Commission [26]. It is a generic study to demonstrate the applicability of the precedingly developed PA tools to a repository in granite. A hypothetical repository site has been selected mainly according to information from performance assessments in Switzerland and Sweden, but also taking into account potential host rock formations in Germany.

The near field is an underground mine consisting of a network of access drifts and boreholes. From each drift a number of boreholes is accessible, each borehole being filled with one container.

Water may access the boreholes via water conducting features in the rock. This water is assumed to flow through the excavation disturbed zone (EDZ) surrounding the borehole and to resaturate the bentonite instantaneously. Release of radionuclides from the container is by diffusion through the bentonite buffer into the EDZ. An advective flow of water through the access galleries is prevented by the bentonite/sand backfill. Relevant input data related to the modelling of near field releases is compiled in TABLE 5-3 (geometry and hydraulics), TABLE 5-6 (waste inventory), and TABLE 5-7 (geochemical data), respectively.

Flow of contaminated water away from the boreholes is via water conducting features, which are modelled as plane fractures. After flow through these fractures, a release into a major fault occurs. It is assumed that transport in the major fault is fast and doesn't need to be calculated. Thus, the far field calculation is based on flow through plane fractures, assuming matrix diffusion into the wall rock. Sorption at the altered wall rock within a specified penetration depth is taken into account. In the upper aquifer, a dilution of radionuclide concentrations is taken into account, which is given by the ratio of volumetric flows in the aquifer and the geosphere. Input data for the far field modelling is given in TABLE 5-4, and sorption data for the far field in TABLE 5-7.

The application of the biosphere model is according to German regulations [27], where the key indicator for long-term safety is the dose rate. The model assumes the usage of contaminated water from a well. The following exposition pathways for human beings are taken into account:

- Drinking water with a consumption rate of 0.8 m^3 per year,
- Ingestion of fish from contaminated ponds,
- Ingestion of plants irrigated with contaminated water,
- Ingestion of milk and meat from cattle whose food has been irrigated with contaminated water,
- Ingestion of milk and meat from cattle maintained with contaminated water,
- External radiation by dwelling on contaminated areas.

A detailed calculation of transport in the biosphere finally results in the biosphere dose conversion factors compiled in TABLE 5-6.

5.3 ENRESA-2000

ENRESA-2000 is a performance assessment exercise for a repository of spent fuel in a generic granitic formation in Spain. The hypothetical repository site has been defined according to information from potential host rock formations in Spain. A compilation of the general data of the repository is given in TABLE 5-2.

At a depth of 500 m two access drifts and 49 disposal drifts are excavated. Disposal containers are emplaced horizontally in the disposal drifts with the help of a carbon steel liner and surrounded by bentonite. The access drifts are backfilled with a 20/80 mixture of bentonite/crushed granite. Data regarding the geometry of the system are given in TABLE 5-3, all other data regarding the near field are given in TABLE 5-6 and TABLE 5-7.

Water reaches the disposal drifts via small fractures. Due to high suction pressures in bentonite it takes about 20 years to fully saturate the bentonite. Minimum container lifetime due to general corrosion is 1300 years and all the containers are assumed to fail 1300 years after disposal. After container failure, and since no credit is given to the cladding as a barrier, water reaches the irradiated UO_2 and radionuclide releases start.

Granite close to the disposal drift is modelled as an equivalent porous medium, with water flow parallel to the disposal drift axis. There is an excavation disturbed zone (EDZ) surrounding the disposal drift where the Darcy velocity is assumed to be 10 times higher than in intact host rock. Radionuclides diffuse radially through the bentonite into the EDZ granite and adjacent intact granite, where flowing water transports the contaminants out of the near field. Near field transport calculations are performed for a single container, and releases from the near field of the total repository are calculated simply multiplying by 3600.

The geosphere has been modelled as an equivalent porous medium with main fractures explicitly represented. Using this model it has been found that the contaminants released from the repository are discharged to a river, and that water travel time from repository to the river is 8400 years.

Since transport in granite is controlled by fractures, in transport calculations the geosphere is modelled as a single one-dimensional planar fracture (or stream tube) with 8400 years water travel time. Longitudinal dispersion and matrix diffusion into the wall rock are modelled, including sorption onto the rock matrix. Neither solubility limits nor sorption on fracture surfaces or infill are considered. General parameters of the far field are given in TABLE 5-4, sorption data are given in TABLE 5-7.

Radionuclides that pass the geosphere are discharged to a river which flows at a rate of $1 \cdot 10^6$ m^3 per year. Radionuclide concentrations in river water are calculated dividing the release rate from the geosphere by the river flow rate.

River water is used by the critical group, which is a self-sufficient community that produces most of its aliments using water from the river. Doses to an average member of the critical group are calculated and compared with a dose limit of $1 \cdot 10^{-4}$ Sv/y. Doses are calculated taking into account a set of exposure pathways similar to that used in SPA-GRS.

Calculations of transport in the biosphere provide the dose conversion factors compiled in TABLE 5-6.

5.4 Kristallin-I

The Kristallin-I study presents a comprehensive description of the post-closure radiological safety assessment of a repository for vitrified high-level radioactive waste (HLW), sited in the crystalline basement of Northern Switzerland [22]. The safety concept for the disposal of HLW in the crystalline basement includes consideration of both engineered and natural geological barriers. It is expected that most radionuclides would decay to insignificant levels within the engineered barriers. The geological barriers provide a stable and protective environment for the engineered barriers, ensuring their longevity; they also have the potential to provide retardation (with consequent radioactive decay) of any radionuclides that eventually escape from the engineered barriers. Their nuclide retention capacity is optimised through the siting of the repository in a low-permeability host rock, with favourable groundwater chemistry in a tectonically stable location at a depth of ca. 1000 meters.

The key characteristics of the disposal system considered in the Kristallin-I performance assessment are summarised in TABLE 5-2. For the total inventory, full reprocessing of all spent fuel arisings from the Swiss 120 GWy(e) scenario (i. e. five nuclear power plants with a total electric capacity of 3 GW, 40 years operation period each) is assumed. The nuclide inventory presented in TABLE 5-6 is an internal update of 1997, based on the COGEMA (France) specification for vitrified HLW issued in 1986.

The engineered barrier system considered in Kristallin-I consists of the vitrified waste form, contained in massive steel containers, surrounded by bentonite and emplaced horizontally in disposal tunnels.

At the presumed time of container failure, 1000 years after repository closure, water contacts the glass, which begins to corrode with an adopted rate of $7.3 \cdot 10^{-5}$ kg/(m^{2}y) (updated value). The glass is assumed to be cracked due to stresses induced during cooling and minor handling shocks.

The diffusion of radionuclides in compacted bentonite is described by an element-specific apparent diffusion constant. The retention factor includes retardation due to chemical interaction (sorption) of the radionuclide with the mineral surfaces in the bentonite. The element-specific solubility limits and the distribution coefficients used to calculate the retention factors for the (bentonite) near field which were adopted in the present work are an 1997 update of the corresponding values of the original Kristallin-I study (see TABLE 5-7).

TABLE 5-3 lists some relevant parameters which form the data base for the calculation of the near field releases. The formalised model STRENG for the near field [28] considers the release of radionuclides from the waste matrix, the diffusive transport of radionuclides through the bentonite, and the release of radionuclides to groundwater at the bentonite-host rock interface.

Actual hydrogeological data and structural information were used to draw up a conceptual model of the crystalline basement of Northern Switzerland. The indications are that the basement can be divided into an upper, higher-permeability domain several hundred metres thick and an underlying low-permeability domain. A repository would be located in the low-permeability domain where groundwater fluxes are expected to be small.

All domains of the crystalline basement contain a series of small-scale discontinuities (water-conducting features) which provide preferential paths for groundwater flow. Some features will inevitably be intersected by the repository tunnel system, thus providing a route by which radionuclides released from the repository near field may be transported to the biosphere. It is assumed that radionuclides reaching the higher-permeability domain are transported rapidly to the biosphere. Fractured aplite/pegmatite dykes and aplitic gneisses with limited matrix diffusion (called 'Geometry 5' - see Table 5.3.4 in [22]) have been identified as the water-conducting features leading to the highest geosphere releases. The corresponding model assumptions and parameter values are summarized in TABLE 5-4. Element-specific distribution coefficients of safety-relevant radionuclides used for geosphere transport calculations are listed in TABLE 5-7.

For the present study, the recently developed computer code PICNIC [29] (instead of the original code RANCHMD [30]) has been applied to calculate the one-dimensional advective-dispersive transport of single radionuclides and decay chains inside the water-conducting features of the geosphere. These features are represented either by sets of planar fractures or cylindrical veins. The programme accounts for the diffusion of radionuclides into the stagnant porewater of the adjacent rock matrix and for the (linear) sorption on the solid phases along the transport path.

The release of radionuclides into the biosphere is most likely to occur somewhere in the upper Rhine valley, where groundwater would mix with recent meteoric and river water in gravel aquifers typical of the valley. A conservative model of human behaviour is imposed such that doses are received from a wide range of exposure pathways, namely ingestion pathways, external γ-irradiation and dust inhalation. A detailed description of the 'Terrestrial - Aquatic Model of the Environment' (TAME) for a wide range of biosphere scenarios representative of possible conditions of exposure is given in [31]. The biosphere dose conversion factors for the conversion of a concentration of contaminated groundwater into a radiation dose are compiled in TABLE 5-6.

5.5 <u>TILA-99</u>

Spent fuel from the Olkiluoto (2 x 840 MWe BWR) and Loviisa (2 x 488 MWe VVER-440 type PWR) nuclear power plants is planned to be disposed of in the repository that Posiva plans to construct in the crystalline bedrock at Olkiluoto. The TILA-99 safety assessment [46] was carried out in association of the Decision in Principle process of the planned disposal facility. The Finnish Parliament ratified in May 2001 the Government's positive Decision in Principle on Posiva's application to locate the spent fuel repository at Olkiluoto.

Two Olkiluoto-specific scenarios of TILA-99 are scrutinised in SPIN:

- SH-sal50: small initial hole (5 mm^2) in the copper-iron container, saline groundwater (as today at the depth of 500 metres), median flow and transport

- DC-ns50: container 'disappearing' at 10 000 years, non-saline groundwater (a possible future situation due the ongoing land uplift), median flow and transport.

All calculations are made for a single container containing 2.14 t_{HM} of BWR fuel.

The conceptual near-field transport models of TILA-99 are presented in FIGURE 5-3. The model for the 'disappearing canister' case consists of five main volumes:
- water volume in the container interior
- bentonite around the container
- bentonite above the container
- backfill of crushed rock and bentonite in the top of the deposition hole
- backfill in the tunnel.

There are three escape routes from the near-field into the geosphere:
- from the bentonite around the container into the rock fissures intersecting the deposition hole (Q_F in FIGURE 5-3)
- from the backfill at the top of the deposition hole into the excavation damaged rock zone (EDZ) below the tunnel floor (Q_{DZ})
- from the tunnel into the rock or EDZ (Q_{TDZ1}).

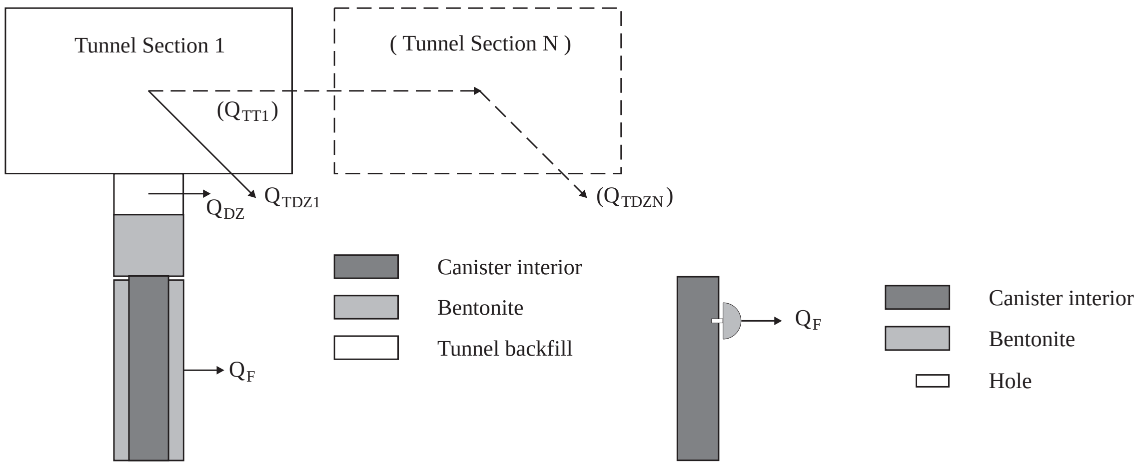

FIGURE 5-3 Conceptual near-field transport models for the 'disappearing canister' (left) and 'small hole' cases (right) of TILA-99

The releases via the three routes are summed up to form the total release rate from the near field into the geosphere, which is then used as the input for the geosphere migration analysis. Transport by advection and diffusion along the tunnel could also be modelled, but in the reference scenarios it is conservatively assumed that radionuclides are released from the first tunnel compartment directly into the geosphere ($Q_{TT1} = 0$). Based on the results of the groundwater flow and transport analyses, the following values have been selected for the transfer coefficients in the DC-ns50 scenario: $Q_F = 0.2$ l/y, $Q_{DZ} = 2$ l/y, $Q_{TDZ1} = 100$ l/y. The resulting equivalent flow rate from the container interior into the geosphere is 3.9 l/y for neutral species, 0.44 l/y for anions and 38 l/y for cations. The difference is due to the different effective diffusivities of the species in buffer and backfill.

In the 'small hole' model only a small fraction of the bentonite in the deposition hole is taken into account, and all flow of groundwater around the deposition hole is assumed to take place in a single fracture just opposite the hole in the container (Q_F = 0.2 l/y). A dominant part of the transport resistance is caused by the small size (5 mm^2) of the hole in the container. The resulting equivalent flow rate from the container interior into the geosphere is in the SH-sal50 scenario 0.0049 l/y for neutral species, 0.0018 l/y for anions and 0.0060 l/y for cations.

The fuel matrix degradation model is the 'alpha-auto-oxidation' model. In addition there are source terms representing instant release fractions and activation products in the zircaloy and other metal parts of the fuel assemblies. Solubility limits (applied only in the container interior) and sorption data in the bentonite buffer are presented in table 5-7. Effective porosities (ε) and diffusivities (D_e) in the buffer are:

- neutral species: $\varepsilon = 0.43$ and $D_e = 1\cdot10^{-10}$ m^2/s in saline and non-saline groundwater
- anions: $\varepsilon = 0.05$ and $D_e = 1\cdot10^{-11}$ m^2/s in saline groundwater, $\varepsilon = 0.05$ and $D_e = 5\cdot10^{-12}$ m^2/s in non-saline groundwater
- cations: $\varepsilon = 0.43$ and $D_e = 1\cdot10^{-9}$ m^2/s in saline groundwater, $\varepsilon = 0.43$ and $D_e = 5\cdot10^{-9}$ m^2/s in non-saline groundwater.

In geosphere transport of radionuclides in fractured rock, a key parameter is the ratio of the flow-wetted fracture surface and the flow rate along the migration path (WL/Q). Based on the groundwater flow and transport analyses, a WL/Q of $5\cdot10^4$ y/m has been selected for the median flow and transport scenarios of TILA-99. With a fracture aperture of $5\cdot10^{-4}$ m and a path length of 600 m, this corresponds to a groundwater transit time of 25 years.

The total penetration depth of matrix diffusion is taken to be 10 cm. The effective diffusivity of non-anionic species is $1\cdot10^{-13}$ m^2/s in the first centimetre of the rock matrix adjacent to the water-conducting fracture, and $1\cdot10^{-14}$ m^2/s between 1 and 10 cm. For anions the effective diffusivity is by a factor 2 (saline groundwater) or 10 (non-saline groundwater) lower than for the non-anionic species. Sorption data are presented in TABLE 5-7.

Dose assessment was based on a simple well scenario where the annual releases from the repository into the biosphere were assumed to be diluted in 100 000 m^3 of water and an individual was assumed to drink 500 litres of contaminated water per year. Biosphere dose conversion factors are presented in TABLE 5-6.

TABLE 5-6 Nuclide-specific data

Radio-nuclide	Half-life [y]	Inventory at time of repository closure [Bq/t_{HM}]				Factor for total activity		Ingestion dose coefficient [Sv/Bq]		Biosphere dose conversion factor [$(Sv:y)/(Bq:m^3)$]			
		SPA-GRS	ENRESA	Kristallin	TILA-99	α	β/γ	ICRP72	IDC-SPIN	SPA-GRS	ENRESA	Kristallin	TILA-99
C-14	$5.733\cdot10^{03}$	$4.57\cdot10^{10}$	$6.35\cdot10^{10}$	-	$3.5\cdot10^{10}$	0	1	$5.80\cdot10^{-10}$	$5.80\cdot10^{-10}$	$1.00\cdot10^{-7}$	$1.55\cdot10^{-8}$	Not used	$2.9\cdot10^{-10}$
Cl-36	$3.012\cdot10^{05}$	$6.10\cdot10^{08}$	$4.79\cdot10^{08}$	-	$1.45\cdot10^{09}$	0	1	$9.30\cdot10^{-10}$	$9.30\cdot10^{-10}$	$2.60\cdot10^{-8}$	$1.87\cdot10^{-9}$	Not used	$4.7\cdot10^{-10}$
Ca-41	$8.100\cdot10^{04}$	$3.00\cdot10^{07}$	$7.67\cdot10^{06}$	-	-	0	1	$1.90\cdot10^{-10}$	$1.90\cdot10^{-10}$	$3.10\cdot10^{-9}$	$3.80\cdot10^{-10}$	Not used	Not used
Ni-59	$8.000\cdot10^{04}$	$5.07\cdot10^{11}$	$1.21\cdot10^{11}$	$1.59\cdot10^{09}$	$2.0\cdot10^{11}$	0	1	$6.30\cdot10^{-11}$	$6.30\cdot10^{-11}$	$1.70\cdot10^{-9}$	$2.82\cdot10^{-10}$	$1.87\cdot10^{-10}$	$3.2\cdot10^{-11}$
Ni-63	$9.200\cdot10^{01}$	$4.59\cdot10^{13}$	$1.41\cdot10^{13}$	$1.54\cdot10^{13}$	$2.7\cdot10^{13}$	0	1	$1.50\cdot10^{-10}$	$1.50\cdot10^{-10}$	$1.10\cdot10^{-9}$	$3.84\cdot10^{-10}$	Not used	$7.5\cdot10^{-11}$
Se-79	$1.100\cdot10^{06}$	$1.10\cdot10^{09}$	$1.04\cdot10^{09}$	$8.65\cdot10^{08}$	$9.3\cdot10^{08}$	0	1	$2.90\cdot10^{-09}$	$2.90\cdot10^{-09}$	$2.30\cdot10^{-7}$	$8.97\cdot10^{-7}$	$4.79\cdot10^{-08}$	$1.5\cdot10^{-9}$
Rb-87	$4.699\cdot10^{10}$	$1.05\cdot10^{06}$	$9.38\cdot10^{05}$	-	-	0	1	$1.50\cdot10^{-09}$	$1.50\cdot10^{-09}$	$1.30\cdot10^{-7}$	$2.15\cdot10^{-8}$	Not used	Not used
Sr-90	$2.914\cdot10^{01}$	$8.99\cdot10^{14}$	$9.54\cdot10^{14}$	** $4.61\cdot10^{14}$	$1.4\cdot10^{15}$	0	2	$2.80\cdot10^{-08}$	$3.07\cdot10^{-08}$	$2.00\cdot10^{-7}$	$6.87\cdot10^{-8}$	Not used	$1.5\cdot10^{-8}$
Zr-93	$1.531\cdot10^{06}$	$9.86\cdot10^{10}$	$9.44\cdot10^{10}$	$6.75\cdot10^{10}$	$9.8\cdot10^{10}$	0	2	$1.10\cdot10^{-09}$	$1.22\cdot10^{-09}$	$6.00\cdot10^{-9}$	$2.36\cdot10^{-9}$	$2.75\cdot10^{-08}$	$6.1\cdot10^{-10}$
Nb-94	$2.031\cdot10^{04}$	$8.49\cdot10^{10}$	$5.81\cdot10^{10}$	** $4.90\cdot10^{08}$	$1.03\cdot10^{10}$	0	1	$1.70\cdot10^{-09}$	$1.70\cdot10^{-09}$	$9.20\cdot10^{-8}$	$9.77\cdot10^{-8}$	Not used	$8.5\cdot10^{-10}$
Mo-93	$3.501\cdot10^{03}$	$4.29\cdot10^{09}$	$1.02\cdot10^{09}$	-	-	0	2	$3.10\cdot10^{-09}$	$3.22\cdot10^{-09}$	$2.80\cdot10^{-8}$	$1.32\cdot10^{-8}$	Not used	Not used
Tc-99	$2.132\cdot10^{05}$	$6.43\cdot10^{11}$	$5.75\cdot10^{11}$	$6.16\cdot10^{11}$	$5.2\cdot10^{11}$	0	1	$6.40\cdot10^{-10}$	$6.40\cdot10^{-10}$	$4.90\cdot10^{-9}$	$1.35\cdot10^{-9}$	$2.04\cdot10^{-08}$	$3.2\cdot10^{-10}$
Pd-107	$6.501\cdot10^{06}$	$5.20\cdot10^{09}$	$5.24\cdot10^{09}$	$4.92\cdot10^{09}$	$4.9\cdot10^{09}$	0	1	$3.70\cdot10^{-11}$	$3.70\cdot10^{-11}$	$3.00\cdot10^{-10}$	$8.14\cdot10^{-9}$	$6.05\cdot10^{-11}$	$1.9\cdot10^{-11}$
Sn-126	$1.001\cdot10^{05}$	$2.79\cdot10^{10}$	$3.49\cdot10^{10}$	$6.50\cdot10^{10}$	$3.2\cdot10^{10}$	0	2.14	$4.70\cdot10^{-09}$	$5.07\cdot10^{-09}$	$8.70\cdot10^{-6}$	$1.97\cdot10^{-7}$	$7.15\cdot10^{-08}$	$2.5\cdot10^{-9}$
I-129	$1.571\cdot10^{07}$	$1.52\cdot10^{09}$	$1.41\cdot10^{09}$	$1.13\cdot10^{06}$	$1.3\cdot10^{09}$	0	1	$1.10\cdot10^{-07}$	$1.10\cdot10^{-07}$	$3.70\cdot10^{-7}$	$1.99\cdot10^{-7}$	$1.13\cdot10^{-06}$	$5.5\cdot10^{-8}$
Cs-135	$2.301\cdot10^{06}$	$1.53\cdot10^{10}$	$2.23\cdot10^{10}$	$1.46\cdot10^{10}$	$2.4\cdot10^{10}$	0	1	$2.00\cdot10^{-09}$	$2.00\cdot10^{-09}$	$8.60\cdot10^{-8}$	$7.50\cdot10^{-8}$	$9.35\cdot10^{-09}$	$1.0\cdot10^{-9}$
Cs-137	$3.002\cdot10^{01}$	$1.34\cdot10^{15}$	$1.44\cdot10^{15}$	** $6.66\cdot10^{14}$	$2.1\cdot10^{15}$	0	2	$1.30\cdot10^{-08}$	$1.30\cdot10^{-08}$	$1.30\cdot10^{-6}$	$3.97\cdot10^{-8}$	Not used	$6.5\cdot10^{-9}$
Sm-147	$1.071\cdot10^{11}$	$6.68\cdot10^{04}$	$1.86\cdot10^{05}$	-	-	1	0	$4.90\cdot10^{-08}$	$4.90\cdot10^{-08}$	$1.60\cdot10^{-7}$	$1.31\cdot10^{-7}$	Not used	Not used
Sm-151	$9.006\cdot10^{01}$	$7.87\cdot10^{12}$	$1.09\cdot10^{13}$	** $8.54\cdot10^{12}$	$1.5\cdot10^{13}$	0	1	$9.80\cdot10^{-11}$	$9.80\cdot10^{-11}$	$3.00\cdot10^{-10}$	$1.70\cdot10^{-10}$	Not used	$4.9\cdot10^{-11}$
Ho-166m	$1.200\cdot10^{03}$	-	$1.42\cdot10^{08}$	** $1.26\cdot10^{08}$	-	0	1	$2.00\cdot10^{-09}$	$2.00\cdot10^{-09}$	Not used	$1.02\cdot10^{-7}$	Not used	Not used
Series 4n:													
Cm-248	$3.393\cdot10^{05}$	$2.91\cdot10^{05}$	$3.88\cdot10^{04}$	-	-	1	0	$7.70\cdot10^{-07}$	$7.70\cdot10^{-07}$	$9.80\cdot10^{-6}$	$2.27\cdot10^{-5}$	Not used	Not used
Pu-244	$8.267\cdot10^{07}$	$4.34\cdot10^{04}$	$2.75\cdot10^{04}$	-	-	1	2	$2.40\cdot10^{-07}$	$2.41\cdot10^{-07}$	$3.40\cdot10^{-6}$	$9.58\cdot10^{-7}$	Not used	Not used
Cm-244	$1.812\cdot10^{01}$	$2.16\cdot10^{14}$	$2.45\cdot10^{13}$	$3.45\cdot10^{12}$	-	1	0	$1.20\cdot10^{-07}$	$1.20\cdot10^{-07}$	$1.40\cdot10^{-6}$	$1.82\cdot10^{-6}$	Not used	Not used
Pu-240	$6.542\cdot10^{03}$	$2.18\cdot10^{13}$	$2.21\cdot10^{13}$	$1.59\cdot10^{11}$	$1.9\cdot10^{13}$	1	0	$2.50\cdot10^{-07}$	$2.50\cdot10^{-07}$	$2.20\cdot10^{-6}$	$8.47\cdot10^{-7}$	$7.15\cdot10^{-07}$	$1.3\cdot10^{-7}$
Np-236 *	$1.540\cdot10^{05}$	$1.40\cdot10^{05}$	$2.62\cdot10^{05}$	-	-	0.09	1	$1.70\cdot10^{-08}$	$2.48\cdot10^{-08}$	$6.20\cdot10^{-6}$	$3.00\cdot10^{-8}$	Not used	Not used
U-236	$2.343\cdot10^{07}$	$1.19\cdot10^{10}$	$1.08\cdot10^{10}$	$1.28\cdot10^{07}$	$1.0\cdot10^{10}$	1	0	$4.70\cdot10^{-08}$	$4.70\cdot10^{-08}$	$2.20\cdot10^{-7}$	$8.05\cdot10^{-8}$	$1.71\cdot10^{-07}$	$2.4\cdot10^{-8}$
Th-232	$1.406\cdot10^{10}$	$3.66\cdot10^{01}$	$2.82\cdot10^{01}$	$6.21\cdot10^{-04}$	-	6	4	$2.30\cdot10^{-07}$	$1.06\cdot10^{-06}$	$1.10\cdot10^{-5}$	$4.77\cdot10^{-6}$	$2.48\cdot10^{-05}$	Not used
U-232	$7.204\cdot10^{01}$	$4.33\cdot10^{08}$	-	$5.28\cdot10^{05}$	-	6	2	$3.30\cdot10^{-07}$	$4.73\cdot10^{-07}$	$5.20\cdot10^{-6}$	In Np-236	Not used	Not used

TABLE 5-6 Nuclide-specific data (continued)

Radio-nuclide	Half-life [y]	Inventory at time of repository closure [Bq/t_{HM}]				Factor for total activity		Ingestion dose coefficient [Sv/Bq]		Biosphere dose conversion factor [(Sv:y)/(Bq:m3)]			
		SPA-GRS	ENRESA	Kristallin	TILA-99	α	β/γ	ICRP72	IDC-SPIN	SPA-GRS	ENRESA	Kristallin	TILA-99
Series 4n+1:													
Cm-245	$8.505\cdot10^{03}$	$1.69\cdot10^{10}$	$1.50\cdot10^{10}$	$4.26\cdot10^{09}$	$1.9\cdot10^{10}$	1	0	$2.10\cdot10^{-07}$	$2.10\cdot10^{-07}$	$3.00\cdot10^{-6}$	$4.07\cdot10^{-6}$	$1.71\cdot10^{-06}$	$1.1\cdot10^{-7}$
Pu-241	$1.441\cdot10^{01}$	$3.23\cdot10^{14}$	$4.83\cdot10^{14}$	-	$1.4\cdot10^{15}$	0	1	$4.80\cdot10^{-09}$	$4.80\cdot10^{-09}$	$4.30\cdot10^{-8}$	$1.22\cdot10^{-8}$	$6.05\cdot10^{-09}$	In Cm-245
Am-241	$4.325\cdot10^{02}$	$1.76\cdot10^{14}$	$1.61\cdot10^{14}$	$2.19\cdot10^{13}$	$1.7\cdot10^{14}$	1	0	$2.00\cdot10^{-07}$	$2.00\cdot10^{-07}$	$2.70\cdot10^{-6}$	$4.50\cdot10^{-7}$	$1.76\cdot10^{-07}$	$1.0\cdot10^{-7}$
Np-237	$2.141\cdot10^{06}$	$1.89\cdot10^{10}$	$1.87\cdot10^{10}$	$1.30\cdot10^{10}$	$1.7\cdot10^{10}$	1	1	$1.10\cdot10^{-07}$	$1.11\cdot10^{-07}$	$6.20\cdot10^{-6}$	$1.94\cdot10^{-7}$	$1.43\cdot10^{-05}$	$5.5\cdot10^{-8}$
U-233	$1.586\cdot10^{05}$	$7.12\cdot10^{06}$	$4.42\cdot10^{06}$	$3.63\cdot10^{06}$	(low)	1	0	$5.10\cdot10^{-08}$	$5.10\cdot10^{-08}$	$2.80\cdot10^{-7}$	$9.08\cdot10^{-8}$	$9.35\cdot10^{-07}$	$2.6\cdot10^{-8}$
Th-229	$7.344\cdot10^{03}$	$1.06\cdot10^{05}$	$2.25\cdot10^{04}$	$3.38\cdot10^{02}$	(low)	5	3	$4.90\cdot10^{-07}$	$6.13\cdot10^{-07}$	$5.40\cdot10^{-6}$	$3.22\cdot10^{-6}$	$1.60\cdot10^{-06}$	$3.1\cdot10^{-7}$
Series 4n+2:													
Cm-246	$4.734\cdot10^{03}$	$4.21\cdot10^{10}$	$3.62\cdot10^{09}$	$7.81\cdot10^{08}$	$3.6\cdot10^{09}$	1	0	$2.10\cdot10^{-07}$	$2.10\cdot10^{-07}$	$2.60\cdot10^{-6}$	$1.82\cdot10^{-6}$	$9.35\cdot10^{-07}$	$1.1\cdot10^{-7}$
Pu-242	$3.872\cdot10^{05}$	$1.15\cdot10^{11}$	$8.47\cdot10^{10}$	$7.78\cdot10^{07}$	$8.1\cdot10^{10}$	1	0	$2.40\cdot10^{-07}$	$2.40\cdot10^{-07}$	$2.10\cdot10^{-6}$	$8.52\cdot10^{-7}$	$1.05\cdot10^{-06}$	$1.2\cdot10^{-7}$
Am-242m	$1.521\cdot10^{02}$	$1.45\cdot10^{11}$	$7.61\cdot10^{11}$	$3.53\cdot10^{11}$	-	0.83	2	$1.90\cdot10^{-07}$	$2.00\cdot10^{-07}$	$2.60\cdot10^{-6}$	$4.50\cdot10^{-7}$	Not used	Not used
Pu-238	$8.780\cdot10^{01}$	$1.02\cdot10^{14}$	$1.10\cdot10^{14}$	$2.62\cdot10^{11}$	$1.3\cdot10^{14}$	1	0	$2.30\cdot10^{-07}$	$2.30\cdot10^{-07}$	$2.00\cdot10^{-6}$	$8.92\cdot10^{-7}$	Not used	$1.2\cdot10^{-7}$
U-238	$4.471\cdot10^{09}$	$1.16\cdot10^{10}$	$1.16\cdot10^{10}$	$1.39\cdot10^{07}$	$1.2\cdot10^{10}$	1	2	$4.50\cdot10^{-08}$	$4.84\cdot10^{-08}$	$3.10\cdot10^{-7}$	$8.32\cdot10^{-8}$	$1.71\cdot10^{-07}$	$2.4\cdot10^{-8}$
U-234	$2.447\cdot10^{05}$	$5.31\cdot10^{10}$	$5.83\cdot10^{10}$	$9.55\cdot10^{07}$	$5.4\cdot10^{10}$	1	0	$4.90\cdot10^{-08}$	$4.90\cdot10^{-08}$	$2.40\cdot10^{-7}$	$8.44\cdot10^{-8}$	$2.26\cdot10^{-07}$	$2.5\cdot10^{-8}$
Th-230	$7.705\cdot10^{04}$	$2.37\cdot10^{07}$	$2.36\cdot10^{07}$	$2.18\cdot10^{06}$	(low)	1	0	$2.10\cdot10^{-07}$	$2.10\cdot10^{-07}$	$2.40\cdot10^{-6}$	$5.29\cdot10^{-6}$	$1.05\cdot10^{-06}$	$1.1\cdot10^{-7}$
Ra-226	$1.601\cdot10^{03}$	$2.89\cdot10^{05}$	$2.56\cdot10^{05}$	$1.01\cdot10^{03}$	(low)	5	4	$2.80\cdot10^{-07}$	$2.17\cdot10^{-06}$	$1.50\cdot10^{-5}$	$3.28\cdot10^{-6}$	$5.50\cdot10^{-07}$	$1.1\cdot10^{-6}$
Series 4n+3:													
Cm-247	$1.561\cdot10^{07}$	$9.36\cdot10^{04}$	$1.27\cdot10^{04}$	-	-	1	1	$1.90\cdot10^{-07}$	$1.90\cdot10^{-07}$	$3.80\cdot10^{-6}$	$7.16\cdot10^{-6}$	Not used	Not used
Am-243	$7.385\cdot10^{03}$	$1.33\cdot10^{12}$	$1.07\cdot10^{12}$	$5.45\cdot10^{11}$	$1.1\cdot10^{12}$	1	1	$2.00\cdot10^{-07}$	$2.01\cdot10^{-07}$	$3.50\cdot10^{-6}$	$6.63\cdot10^{-7}$	$8.25\cdot10^{-07}$	$1.0\cdot10^{-7}$
Pu-239	$2.408\cdot10^{04}$	$1.32\cdot10^{13}$	$1.36\cdot10^{13}$	$1.62\cdot10^{10}$	$1.3\cdot10^{13}$	1	1	$2.50\cdot10^{-07}$	$2.50\cdot10^{-07}$	$2.20\cdot10^{-6}$	$8.92\cdot10^{-7}$	$9.35\cdot10^{-07}$	$1.3\cdot10^{-7}$
U-235	$7.043\cdot10^{08}$	$4.86\cdot10^{08}$	$5.37\cdot10^{08}$	$8.80\cdot10^{05}$	$6.7\cdot10^{08}$	1	1	$4.70\cdot10^{-08}$	$4.73\cdot10^{-08}$	$9.40\cdot10^{-7}$	$8.33\cdot10^{-8}$	$1.32\cdot10^{-05}$	$2.4\cdot10^{-8}$
Pa-231	$3.279\cdot10^{04}$	$1.85\cdot10^{06}$	$1.49\cdot10^{06}$	$6.01\cdot10^{05}$	(low)	6	3	$7.10\cdot10^{-07}$	$1.92\cdot10^{-06}$	$1.30\cdot10^{-5}$	$7.26\cdot10^{-6}$	$8.25\cdot10^{-05}$	$9.6\cdot10^{-7}$

(*) Np-236 → U-236 (91%); Np-236 → Pu-236 → U-232 (9%)

(**) Not considered in the calculation of safety and performance indicators, because screening calculations showed a negligible impact on the long-term safety.

TABLE 5-7 Selected element-specific data

Ele-ment	Solubility limits near field [mol/l]					Distribution coefficients near field [m^3/kg]					Distribution coefficients far field [m^3/kg]				
	SPA-GRS	ENRESA	Kristallin	TILA-99 saline	TILA-99 non-saline	SPA-GRS	ENRESA	Kristallin	TILA-99 saline	TILA-99 non-saline	SPA-GRS	ENRESA	Kristallin	TILA-99 saline	TILA-99 non-saline
C	High	High	Not used	High	High	$1.0\cdot10^{-2}$	0	Not used	0	0	$1.0\cdot10^{-3}$	$3.0\cdot10^{-5}$	Not used	$1\cdot10^{-4}$	$1\cdot10^{-4}$
Cl	High	High	Not used	High	High	0	0	Not used	0	0	0	0	Not used	0	0
Ni	High	$1.0\cdot10^{-3}$	High	$1\cdot10^{-4}$	$1.0\cdot10^{-4}$	1.0	0.02	1.0	$1\cdot10^{-3}$	$5\cdot10^{-2}$	$5.0\cdot10^{-1}$	$6\cdot10^{-3}$	$5.0\cdot10^{-1}$	$5\cdot10^{-3}$	$1\cdot10^{-1}$
Se	$1.0\cdot10^{-8}$	$1.0\cdot10^{-8}$	$8.0\cdot10^{-7}$	$1\cdot10^{-6}$	$1.0\cdot10^{-6}$	$5.0\cdot10^{-3}$	0.0005	$5.0\cdot10^{-3}$	0	0	$1.0\cdot10^{-2}$	$3.0\cdot10^{-5}$	$1.0\cdot10^{-2}$	$1\cdot10^{-4}$	$5\cdot10^{-4}$
Sr	$1.0\cdot10^{-5}$	$1.0\cdot10^{-5}$	Not used	$1\cdot10^{-3}$	$1.0\cdot10^{-5}$	$1.0\cdot10^{-2}$	0.005	Not used	$1\cdot10^{-3}$	$5\cdot10^{-2}$	$1.0\cdot10^{-2}$	$3.0\cdot10^{-4}$	Not used	$1\cdot10^{-4}$	$5\cdot10^{-3}$
Zr	$5.0\cdot10^{-9}$	$3.0\cdot10^{-9}$	$5.0\cdot10^{-9}$	$5\cdot10^{-8}$	$5.0\cdot10^{-8}$	1.0	1	1.0	$2\cdot10^{-1}$	$2\cdot10^{-1}$	1.0	$3\cdot10^{-2}$	1.0	$2\cdot10^{-1}$	$2\cdot10^{-1}$
Tc	$1.0\cdot10^{-7}$	$1.0\cdot10^{-8}$	$1.0\cdot10^{-7}$	$5\cdot10^{-8}$	$5.0\cdot10^{-8}$	$1.0\cdot10^{-1}$	0.1	$1.0\cdot10^{-1}$	$1\cdot10^{-2}$	$1\cdot10^{-2}$	$5.0\cdot10^{-1}$	$3.0\cdot10^{-3}$	$1.0\cdot10^{-1}$	$5\cdot10^{-2}$	$5\cdot10^{-2}$
Pd	$1.0\cdot10^{-11}$	$4.0\cdot10^{-9}$	$1.0\cdot10^{-11}$	$1\cdot10^{-8}$	$1.0\cdot10^{-8}$	1.0	0.1	1.0	$1\cdot10^{-3}$	$1\cdot10^{-3}$	$5.0\cdot10^{-1}$	$3.0\cdot10^{-4}$	$5.0\cdot10^{-1}$	$1\cdot10^{-4}$	$1\cdot10^{-3}$
Sn	$1.0\cdot10^{-5}$	$1.0\cdot10^{-8}$	$1.0\cdot10^{-5}$	$5\cdot10^{-6}$	$5.0\cdot10^{-6}$	1.0	1	1.0	$1\cdot10^{-3}$	$1\cdot10^{-3}$	$5.0\cdot10^{-1}$	$3.0\cdot10^{-4}$	$1.0\cdot10^{-1}$	$1\cdot10^{-4}$	$1\cdot10^{-3}$
I	High	High	High	High	High	$5.0\cdot10^{-3}$	0	0.0	0	0	$1.0\cdot10^{-3}$	0	$8.0\cdot10^{-3}$	0	0
Cs	High	High	High	High	High	$1.0\cdot10^{-2}$	0.05	$1.0\cdot10^{-2}$	$4\cdot10^{-2}$	$2\cdot10^{-1}$	$4.2\cdot10^{-2}$	$3.0\cdot10^{-3}$	$4.2\cdot10^{-2}$	$1\cdot10^{-2}$	$5\cdot10^{-2}$
Ra	$1.0\cdot10^{-6}$	$1.0\cdot10^{-7}$	$1.0\cdot10^{-10}$	$1\cdot10^{-6}$	$1\cdot10^{-7}$	$1.0\cdot10^{-2}$	0.005	$1.0\cdot10^{-2}$	$2\cdot10^{-3}$	$1\cdot10^{-1}$	$5.0\cdot10^{-1}$	$9.0\cdot10^{-3}$	$5.0\cdot10^{-1}$	$2\cdot10^{-2}$	$2\cdot10^{-1}$
U	$1.0\cdot10^{-7}$	$5.0\cdot10^{-7}$	$1.0\cdot10^{-7}$	$3\cdot10^{-7}$	$3\cdot10^{-7}$	5.0	0.5	1.0	$5\cdot10^{-2}$	$5\cdot10^{-2}$	1.0	$1.5\cdot10^{-2}$	1.0	$1\cdot10^{-1}$	$1\cdot10^{-1}$
Am	$1.0\cdot10^{-5}$	$1.0\cdot10^{-5}$	$1.0\cdot10^{-5}$	$5\cdot10^{-7}$	$5\cdot10^{-7}$	5.0	3	5.0	$3\cdot10^{-1}$	$3\cdot10^{-1}$	5.0	$1.5\cdot10^{-2}$	2.0	$4\cdot10^{-2}$	$4\cdot10^{-2}$
Cm	$1.0\cdot10^{-5}$	$1.0\cdot10^{-6}$	$1.0\cdot10^{-7}$	$5\cdot10^{-8}$	$5\cdot10^{-8}$	5.0	3	5.0	$3\cdot10^{-1}$	$3\cdot10^{-1}$	5.0	$1.5\cdot10^{-2}$	2.0	$4\cdot10^{-2}$	$4\cdot10^{-2}$
Pu	$1.0\cdot10^{-8}$	$1.0\cdot10^{-8}$	$1.0\cdot10^{-8}$	$5\cdot10^{-7}$	$5\cdot10^{-7}$	5.0	3	5.0	$3\cdot10^{-1}$	$3\cdot10^{-1}$	5.0	$3.0\cdot10^{-2}$	5.0	$5\cdot10^{-1}$	$5\cdot10^{-1}$
Np	$1.0\cdot10^{-10}$	$1.0\cdot10^{-9}$	$5.0\cdot10^{-9}$	$5\cdot10^{-8}$	$5\cdot10^{-8}$	5.0	1	5.0	$1\cdot10^{-1}$	$1\cdot10^{-1}$	1.0	$1.5\cdot10^{-2}$	1.0	$2\cdot10^{-1}$	$2\cdot10^{-1}$
Th	$5.0\cdot10^{-9}$	$1.0\cdot10^{-7}$	$5.0\cdot10^{-9}$	$5\cdot10^{-7}$	$5\cdot10^{-7}$	5.0	3	5.0	$3\cdot10^{-1}$	$3\cdot10^{-1}$	1.0	$1.5\cdot10^{-2}$	1.0	$2\cdot10^{-1}$	$2\cdot10^{-1}$
Pa	$1.0\cdot10^{-10}$	$3.0\cdot10^{-7}$	$1.0\cdot10^{-10}$	$1\cdot10^{-8}$	$1\cdot10^{-8}$	1.0	0.1	1.0	$5\cdot10^{-2}$	$5\cdot10^{-2}$	1.0	$6.0\cdot10^{-3}$	1.0	$5\cdot10^{-2}$	$5\cdot10^{-2}$

6 REFERENCE VALUES FOR SAFETY INDICATORS

To evaluate the calculated results for safety indicators, the end points of the safety assessment must be compared with established reference data that indicate an acceptable level of safety. An objective of the project was to develop indicative reference values for the safety indicators tested. For dose rates, reference values have already been developed in a number of countries. For indicators relating to the amounts of radioactivity reaching the surface environment, the general approach was to develop reference values based on natural radioactivity levels in groundwater. That is, if the level of radioactivity in the environment originating from the repository can be shown to be small compared with the levels of natural radioactivity already present in the environment this supports the argument that the repository is safe.

The reference data are taken from open literature and from documents made available by project participants and from the IAEA Co-ordinated Research Programme 'The use of selected safety indicators in the assessment of radioactive waste disposal'. These data were analysed to develop indicative reference values for the safety indicators, though a statistical analysis of the data was not possible because of differences in how the data were presented in the source documents.

This chapter gives the basis for the reference values for:

- Effective dose rate [Sv/y],
- Radiotoxicity concentration in biosphere water [Sv/m^3],
- Radiotoxicity flux from geosphere [Sv/y],
- Time-integrated radiotoxicity flux from geosphere [Sv],
- Radiotoxicity outside geosphere [Sv],
- Relative activity flux from geosphere.

The reference values presented in this chapter are based on a limited set of data. They are only intended for use as indicative values for the SPIN project and are not proposed for more general use as a basis for regulatory constraints or limits.

6.1 Methodology

6.1.1 Effective dose rate

The *effective dose rate* to exposed individuals is currently the main internationally accepted measure to determine the acceptability of nuclear practices, being the measure on which ICRP recommendations are based. This measure is usually reflected in nuclear safety legislation as an annual exposure limit for members of the public from all such practices, i.e. from activities involving the use of artificial sources of radiation or from natural materials processed because of their radioactive properties. Following ICRP 60 the overall limit is normally 1 mSv/y, representing an appropriately small level of risk to human health. Regulatory authorities often ensure this overall limit is achieved by establishing constraints from specific radiation sources. The *effective dose rate* therefore represents the baseline safety indicator against which the usefulness of other indicators can be judged.

The approach taken to deciding a reference value for *effective dose rate* was to survey the countries participating in SPIN to determine where constraint levels have been established for deep disposal of radioactive waste and to derive reference values from these. Information was also obtained on levels of naturally occurring radiation in participating countries in order that the regulatory constraint values could be seen in the context of the normal variability of natural radiation. In principle, this information could also be used to develop reference values for comparison. This approach has not been taken here because of the potential conflict with constraint values established by the regulatory authorities.

6.1.2 Radiotoxicity fluxes and concentrations

Assumptions

Reference values for *radiotoxicity concentration in biosphere water* and *radiotoxicity flux from geosphere* were determined using the following main assumptions:

- reference values were determined only for biosphere water since in the performance assessments the main exposure paths are by way of biosphere water;
- the major contributors to natural terrestrial radiation are the natural series decay chains headed by U and Th and the non-series nuclides K-40 and Rb-87;
- the U and Th chains are in secular equilibrium;
- only the longer-lived daughters (half-life > 1 day) are mobilised from the geosphere and are released into the biosphere; and
- the calculation of radiotoxicity from activity can be performed using the ingestion dose coefficient (IDC), thus neglecting other pathways.

The above assumptions are considered reasonable as a first approximation. However, it is known from measurements that the radionuclide composition of groundwater can vary to a large extent and certain radionuclides are often not in equilibrium with their parent nuclides, e.g. Ra-226 in the U-238 decay chain.

Radiotoxicity concentration in biosphere water

The following steps have been followed to calculate the total natural radiotoxicity concentration in the relevant biosphere compartment (water):

- Determine mass concentrations (g/m^3) of elemental uranium, thorium, potassium and rubidium.

- Determine relative abundance and specific activities (Bq/g) of different isotopes of uranium and thorium and of potassium and rubidium. For example:U-235 activity = mass of elemental U x isotopic abundance (0,0072) x specific activity of U-235.

- The final step is to convert nuclide activities to radiotoxicity values using the ingestion dose coefficients (IDCs). The total radiotoxicity of. a chain is calculated by summing the values of all its members. It is assumed that each daughter nuclide is in secular equilibrium with its parent. The radiotoxicity values of the chains and single nuclides under consideration are then summed to produce the reference value for the total radiotoxicity.

Radiotoxicity flux from geosphere

The same basic calculational methodology is used, on the basis that fluxes are essentially mobile concentrations. Therefore the calculation is largely the same except that elemental fluxes are required in the initial step instead of elemental concentrations. The final step in this case gives a total radiotoxicity flux at the relevant geosphere/biosphere interface.

Cumulated fluxes

Reference values for *time-integrated radiotoxicity flux from geosphere* and *radiotoxicity outside geosphere* are derived from the reference values for *radiotoxicity flux from geosphere*, on the basis that these indicators are themselves derived from the total radiotoxicity fluxes from the geosphere.

Reference values for *radiotoxicity outside geosphere* could also be obtained by considering the inventories in different compartments of the biosphere. There are several compartments like river sediments and sea waters which could be considered. But those compartments are not uniquely defined and the appropriate dimensions of these compartments and the safety-relevance of their radionuclide inventory is open to debate.

The approach taken in this project is therefore to integrate the reference values determined for radiotoxicity fluxes over the time period of interest.

6.1.3 Relative activity flux from geosphere

This indicator needs a complete set of reference values for all nuclides, to be used as weighting factors, instead of just one single value. For the indicator test a set of values provided by the Finnish regulation authority has been used.

6.2 Determination of reference values

6.2.1 Effective dose rate

The average radiation dose, worldwide, from natural sources is 2.4 mSv/y for adults. This dose level comprises 0.9 mSv/y external exposure (cosmic rays and terrestrial gamma rays) and 1.5 mSv/y internal exposure (mainly from inhalation of radon and from ingestion). Within this average figure there are significant variations depending on location and altitude. The following table of average radiation doses from natural sources is taken from [32]. In general the variation in radiation levels from natural sources is significantly larger than the total exposure from man-made sources.

TABLE 6-1 Annual effective radiation doses from natural sources [mSv/y]

Source	Worldwide average	Typical range	Extremes
External exposure - Cosmic rays - Terrestrial gamma rays	 0.4 0.5	 0.3 - 1 [(a)] 0.3 - 0.6 [(b)]	 2 4.3
Internal exposure - Inhalation (mainly Rn) - Ingestion	 1.2 0.3	 0.2 - 10 [(c)] 0.2 - 0.8 [(d)]	 500
Total	2.4	1 - 10	

(a) Range from sea level to high ground elevations
(b) Depending on radionuclides composition of soil and building material
(c) Depending on indoor accumulation of radon gas
(d) Depending on composition of foods and drinking water

In many countries the regulatory authorities have established constraints on exposure from single sources of radiation, including from disposal facilities, to ensure that the overall limit will be achieved. The reference values for *effective dose rate* are derived from these regulatory constraints. Radiation limits for population are based mainly on ICRP 60 recommendations which formed the basis for the IAEA International Basic Safety Standards and (in the European Union) Council Directive 96/29/Euratom. In accordance with these documents the total effective dose to members of the public should be limited to 1 mSv per year from 'man-made' sources of radiation. On that basis constraints are established on exposure from a single radiation source such as a disposal facility, e.g. a maximum constraint value of 0.3 mSv per year. In ICRP 81 [33] it is stated that "assessed doses or risks ... should be compared with a dose constraint of 0.3 mSv per year ..."

Some of the countries surveyed have established constraint values for geological disposal facilities. The main participant countries where such constraints exist are Finland, Germany, Spain and Switzerland. A constraint value has also been established for potential geological disposal at the Yucca Mountain site in Nevada. For all these cases the constraint value on the effective dose to a member of the public is in the range 0.1 mSv per year (Finland, Spain and Switzerland) to 0.3 mSv/y (Germany). The constraint value for Yucca Mountain is 0.15 mSv/y. On that basis this range of values (0.1 - 0.3 mSv/y) was adopted for the indicator test.

Both the average natural background radiation and its variation are significantly greater than the adopted reverence values.

6.2.2 Radiotoxicity concentration in biosphere water

To obtain reference values for *radiotoxicity concentration in biosphere water* [Sv/m^3], data from measurements have been taken from various sources. The results of these measurements are commonly reported as elemental concentrations [mg/l] or activity concentrations [Bq/l], and have been converted into the unit Sv/m^3 using the ingestion dose coefficients (IDC). See TABLE 6-2.

TABLE 6-2 Radionuclide-specific data

Nuclide	Isotopic abundance [weight %]	Specific activity [Bq/g]	IDC [Sv/Bq]
K-40	0.0118	$2.59 \cdot 10^{5}$	$6.2 \cdot 10^{-9}$
Rb-87	27.85	$3.25 \cdot 10^{3}$	$1.50 \cdot 10^{-9}$
Th-232	100	$4.07 \cdot 10^{3}$	$2.30 \cdot 10^{-7}$
U-235	0.711	$8.00 \cdot 10^{4}$	$4.73 \cdot 10^{-8}$
U-238	99.28	$1.24 \cdot 10^{4}$	$4.50 \cdot 10^{-8}$
Th-232+			$1.06 \cdot 10^{-6}$
U-235+			$1.97 \cdot 10^{-6}$
U-238+			$2.48 \cdot 10^{-6}$

The results of the search for radiotoxicity concentrations in groundwater cover a range of more than 3 orders of magnitude. This applies not only for the ratio between minimum and maximum values for a single set of data, but also for the differences between the average values calculated for a single radionuclide or nuclide chain. From the radiotoxicity concentrations found in the search only a limited dataset is used. The data selected are assumed to relate the best to conditions in the vicinity of potential deep underground repository sites in granite. These data (see TABLE 6-3 and TABLE 6-4) show a smaller range than the full dataset.

In Finland a large amount of groundwater research has been performed. Sampling has been done via boreholes in deep groundwater ('drilled wells') and shallow groundwater ('dug wells'). In Czech Republic, as well as in the Glattfelden test area of Switzerland, concentrations of Th and Rb were not measured. The measured data from Switzerland, Finland, and the Czech Republic are presented in TABLE 6-3 and TABLE 6-4.

TABLE 6-3 Elemental and activity compositions: Switzerland

	K-40 [Bq/m³]	Rb-87 [Bq/m³]	Th232[Bq/m³]	U [µg/l]
Deep groundwater, Northern Switzerland				
Minimum	33	45	16	0.2
Maximum	4800	1154	155	15
Mean value	1516	309	58.4	4.2
Near-surface groundwaters, Glattfelden test area				
Minimum	136			1.1
Maximum	156			1.2
Mean value	146			1.2

TABLE 6-4 Elemental compositions [µg/l]: Finland and Czech Republic

	K	Rb	Th	U
Deep groundwater Finland				
Minimum	230	0.03	<0.02	<0.01
Maximum	40 200	42.7	1.41	643
Median	3000	1.87	<0.02	0.68
Shallow groundwater Finland				
Minimum	190	0.04	<0.02	<0.01
Maximum	92 300	73.3	1.50	36.6
Median	2790	2.73	<0.02	0.01
Shallow groundwater (granitic regions) Czech Republic				
Minimum	400			0.1
Maximum	78 300			20
Median	2900			15

The calculated radiotoxicity concentrations are presented in FIGURE 6-1 in graphical form, for key radionuclides and series and different groundwater types, based on source information from Finland (FI), Switzerland (CH) and Czech Republic (CZ). For each groundwater type the range and mean values, where available, are shown.

For each nuclide or chain an average value (on a logarithmic scale) of the mean values has been calculated. These averages were then aggregated to produce an indicative reference value for the radiotoxicity concentration in biosphere water: $2.0 \cdot 10^{-5}$ Sv/m^3. See TABLE 6-5.

TABLE 6-5 Reference value for *radiotoxicity concentration in biosphere water*

Nuclide (or chain)	Radiotoxicity concentration [Sv/m^3]
K-40	$1.1 \cdot 10^{-6}$
Rb-87	$1.6 \cdot 10^{-8}$
Th-232+	$7.7 \cdot 10^{-7}$
U-235+	$6.2 \cdot 10^{-7}$
U-238+	$1.7 \cdot 10^{-5}$
Sum	$2.0 \cdot 10^{-5}$

The project did not calculate a range for this reference value because the source data did not allow any meaningful conclusions to be drawn about the statistical significance of such a range. Nonetheless it is likely that the range will be at least an order of magnitude on either side of the calculated value.

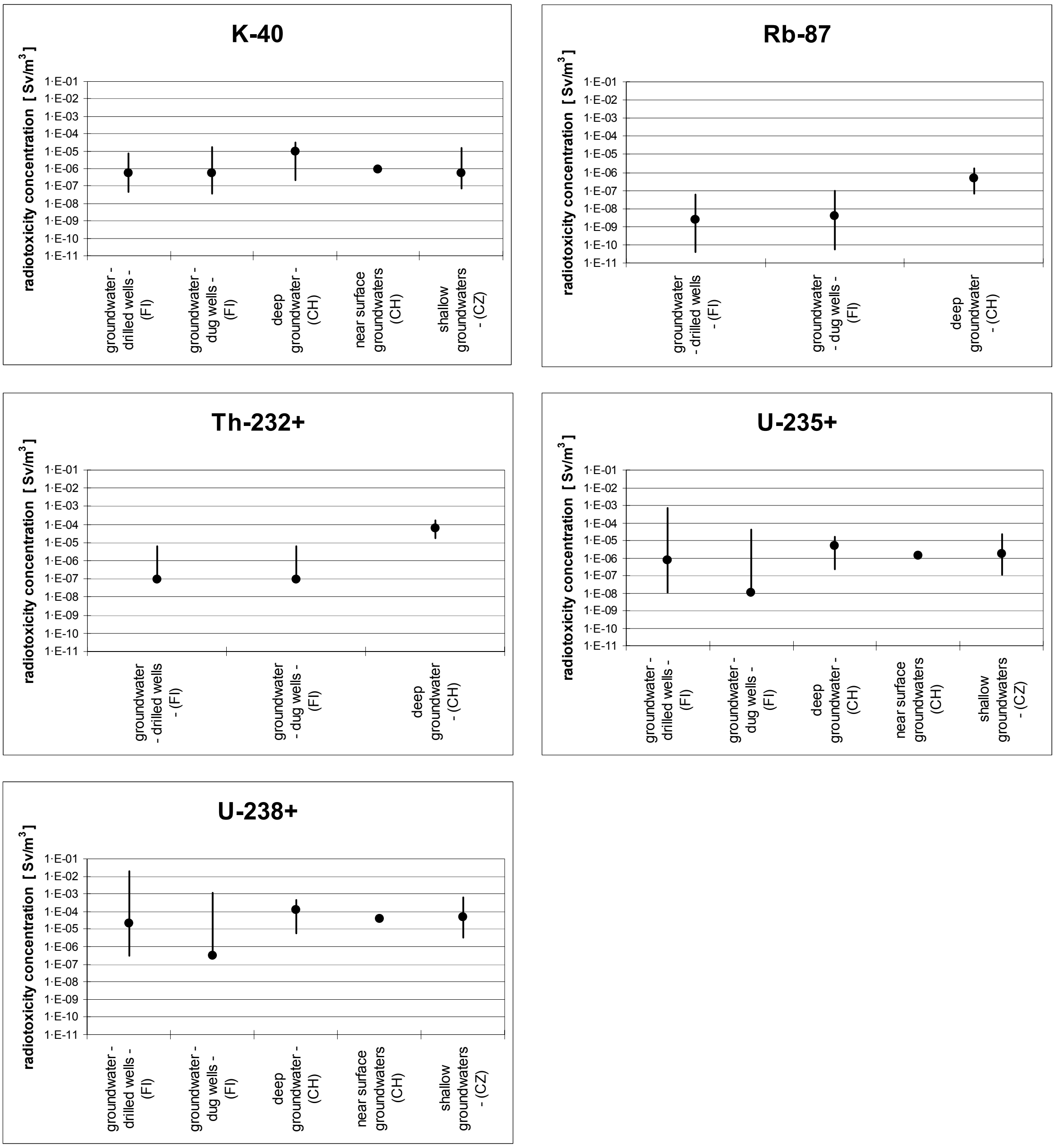

FIGURE 6-1 Radiotoxicity concentrations of key radionuclides and decay chains

6.2.3 Radiotoxicity flux from geosphere

Fluxes of naturally occurring materials are brought about by the action of a number of processes in the surface and subsurface environments, e.g. rock-water interactions, groundwater flow, weathering and erosion, and sediment transport (e.g. by water, wind or ice). Here, only the fluxes resulting from groundwater flow are considered, to allow comparison with the calculated fluxes of radionuclides in the performance assessments included within the project.

Currently available data on elemental or activity fluxes are insufficient to develop a comprehensive range of reference values against which repository-derived fluxes can be compared. The calculations reported use averaged data reported by Miller *et al.* [34] for two potential host environments – an inland pluton and crystalline basement rock under sedimentary cover – and assume temperate climatic conditions. A repository-equivalent surface area of 200 km^2 is assumed in order to calculate total annual fluxes. Total elemental fluxes are converted to radiotoxicity fluxes using the procedure described in section 6.1.

Calculated average elemental fluxes for a temperate climate are given in TABLE 6-6.

TABLE 6-6 Natural elemental fluxes [34]

	Inland Pluton [kg/y]	Crystalline rock/cover [kg/y]
K	$4.5 \cdot 10^3$	$1.6 \cdot 10^4$
Rb	38	$1.4 \cdot 10^2$
Th	0.23	0.81
U	0.76	2.7

These fluxes are used to calculate activity and radiotoxicity fluxes according to the approach set out in the methodology section 6.1. The corresponding values are given in TABLE 6-7.

TABLE 6-7 Natural activity and radiotoxicity fluxes

	Activity flux [Bq/y]		Radiotoxicity flux [Sv/y]	
	Inland pluton	Rock/cover	Inland pluton	Rock/cover
K-40	$1.4 \cdot 10^8$	$4.9 \cdot 10^8$	$8.5 \cdot 10^{-1}$	3.0
Rb-87	$3.4 \cdot 10^7$	$1.2 \cdot 10^8$	$5.1 \cdot 10^{-2}$	0.2
Th-232	$9.3 \cdot 10^5$	$3.3 \cdot 10^6$	$9.9 \cdot 10^{-1}$ *)	3.5 *)
U-235	$4.3 \cdot 10^5$	$1.5 \cdot 10^6$	$8.5 \cdot 10^{-1}$ *)	3.0 *)
U-238	$1.9 \cdot 10^7$	$6.7 \cdot 10^7$	$2.3 \cdot 10^1$ *)	83 *)
Total			26 *)	93 *)

*) Including longer-lived daughters (half-life > 1 day)

Using these data a total radiotoxicity flux of 26 Sv/y was calculated for an inland pluton and a total radiotoxicity flux of 93 Sv/y was calculated for the crystalline basement environment under sedimentary cover. The average of these values was used to determine an indicative reference value of 60 Sv/y for a typical area of 200 km^2. This value is used for comparison with the calculated radiotoxicity fluxes.

6.2.4 Time-integrated radiotoxicity flux from geosphere

This indicator is calculated from *radiotoxicity flux from geosphere* by means of integrating over time. The reference value is calculated accordingly. Since the reference value for *radiotoxicity flux from geosphere* is a constant value, the time integration can be simplified to the following formula for calculation of the reference value: 60 Sv/y · time.

6.2.5 Radiotoxicity outside geosphere

Since natural releases from the geosphere have been calculated assuming secular equilibrium of the members of the decay chains, secular equilibrium will remain valid in the biosphere. Due to their extremely long half-lives, radioactive decay of the parents of the decay chains, Rb-87 and K-40 is negligible during the calculation period ($1 \cdot 10^7$ years). As a consequence, radioactive decay/ingrowth in the biosphere can be neglected, and the reference value for *radiotoxicity outside geosphere* can be taken equal to the reference value for *radiotoxicity flux from geosphere*.

6.2.6 Relative activity flux from geosphere

In the SPIN test calculations, the release rate constraints specified in the Finnish regulatory Guide YVL 8.4 [7] were used as examples of reference values for release rates from the geosphere to the biosphere. However, it should be noted that the Finnish regulator has derived these constraints partly based on a set of reference biospheres considered possible in the future at the planned disposal site, and partly on natural fluxes of radionuclides established for similar environments. The reference values of the Finnish regulatory guide are thus not directly applicable for other disposal concepts and sites.

The release rate constraints of Guide YVL 8.4 consider total repository-induced releases from the geosphere to the biosphere. They are intended to be applied to time frames of several thousands of years while dose rate constraints are applied in the shorter term. The nuclide-specific constraints are defined for long-lived radionuclides only. The effects of their short-lived daughters have been taken into consideration in the constraints defined for the long-lived parents. The nuclide-specific release rate constraints are

- 0.03 GBq/y for the long-lived α-emitting isotopes of Ra, Th, Pa, Pu, Am, Cm
- 0.1 GBq/y for Se-79, I-129, and Np-237
- 0.3 GBq/y for C-14, Cl-36, Cs-135, and the long-lived isotopes of U
- 1 GBq/y for Nb-94 and Sn-126
- 3 GBq/y for Tc-99
- 10 GBq/y for Zr-93
- 30 GBq/y for Ni-59
- 100 GBq/y for Pd-107 and Sm-151.

The release rates can be averaged over 1000 years at the most. The sum of the ratios between the nuclide-specific activity releases and the respective constraints shall be less than one. For the long-lived Ca-41 and Mo-93, for which constraints have not been specified in Guide YVL 8.4, VTT has recommended, respectively, constraints of 10 GBq/y and 0.3 GBq/y for use in SPIN.

6.3 Conclusions

For indicators relating to the amounts of radioactivity reaching the surface environment, the general approach was to develop reference values based on natural radioactivity levels on the basis that small changes to natural levels of radioactivity will be insignificant compared to variations that occur naturally over space and time.

- For *effective dose rate* the established regulatory constraints are used as a basis for comparison. This suggests a 'reference band' between 0.1 mSv/y and 0.3 mSv/y.

- For *radiotoxicity concentration in the biosphere water* an indicative reference value of $2 \cdot 10^{-5}$ Sv/m^3 is used.

- For *radiotoxicity flux from the geosphere* an indicative reference value of 60 Sv/y is used.

- Reference values for *time-integrated radiotoxicity flux from geosphere* and for *radiotoxicity outside geosphere* can be calculated according to: 60 Sv/y · time.

The radionuclides in the Uranium-238 decay chain are in disequilibrium in groundwater but, as a first approximation, the assumption of secular equilibrium provides a reasonable basis for calculating reference values for comparison with the results of repository performance assessments.

A reference value for a safety indicator is only meaningful when it is safety-relevant. Therefore a safety-relevant weighting scheme (e.g. the IDC's) must be used. The only safety indicator which is a direct measure for safety of human beings is the dose. Radiotoxicity concentrations in water do not take account of the exposure pathways for an individual and therefore the relation to safety is less obvious.

Safety indicators based on radionuclide fluxes in the geosphere are less direct measures of safety than concentrations because the effect of the biosphere is not taken into account. For example, the radiotoxicity flux as indicator excludes the uncertainties of the aquifer dilution which derive from the uncertainties of the hydrogeological situation. However, the development of the corresponding reference value is likely to take the hydrogeological situation into account since fluxes can not be measured directly and are derived from measured radionuclide or elemental concentrations and (assumptions about) the groundwater flow.

Because of the large variation in natural elemental abundances, groundwater flows and transport mechanisms, global reference values are likely to be used in conjunction with reference values more representative of the repository site being assessed. The former give an indication of the impact of a repository in terms of values for a particular type of environment, whereas the latter indicate the impact on terms of the undisturbed conditions at a particular site.

The reference values derived in the SPIN project are based on a limited set of data. They are only intended for use as indicative values within this SPIN project and are not proposed for use as the basis for constraints or limits as such values should take into account the specific characteristics of the host environment of the proposed repository.

7 DISCUSSION OF SAFETY INDICATOR RESULTS

In this chapter, the results of the calculations obtained for the selected safety indicators are presented and discussed. The safety indicators are graphically depicted in FIGURE 7-1 to FIGURE 7-6 The horizontal lines in the lower part of each figure indicate the radionuclides with the highest contribution in the corresponding time interval. The results of the calculations are compared with the reference values which have been developed in Chapter 6. These values are compiled in TABLE 7-1.

TABLE 7-1 Reference values for safety indicators as developed for the SPIN project

Safety indicator name	Reference value
Effective dose rate	10^{-4} - $3 \cdot 10^{-4}$ Sv/y
Radiotoxicity concentration in biosphere water	$2 \cdot 10^{-5}$ Sv/m^3
Radiotoxicity flux from geosphere	60 Sv/y
Time-integrated radiotoxicity flux from geosphere	60 Sv/y·time
Radiotoxicity outside geosphere	60 Sv/y·time
Relative activity flux from geosphere	1 (unity)

All safety indicator results depend directly on the considered waste form (spent fuel in the case of ENRESA-2000, SPA-GRS and TILA-99 and vitrified high-level waste for Kristallin-I) and the amount of waste the calculations are based upon. The corresponding values are: 6640 t_{HM} for ENRESA-2000 and 6250 t_{HM} for SPA-GRS (corresponding to 1/4 of the disposed waste). In contrast, calculations for TILA-99 deal with releases from a single defective container containing 2.14 t_{HM}; consequently, for any safety indicator, TILA-99 results are roughly 3 orders of magnitude smaller than those obtained in the case of ENRESA-2000 and SPA-GRS. The vitrified high-level waste assumed in Nagra's Kristallin-I waste inventory results from the reprocessing of 3730 t_{HM} (i.e. half the inventory assumed by Enresa and GRS in their studies).

The results for *effective dose rate*, *radiotoxicity concentration in biosphere water* and *relative activity concentration in biosphere water* depend also on the flow rate of the diluting water bodies (surface water, aquifers) in the biosphere. The assumption within ENRESA-2000 for the diluting river is 10^6 m^3/y, the SPA-GRS study adopts an aquifer flow of $8.0 \cdot 10^6$ m^3/y, Kristallin-I an aquifer flow of $5.5 \cdot 10^6$ m^3/y, and in TILA-99 a total dilution of 10^5 m^3/y in the geosphere and biosphere is assumed.

7.1 Effective dose rate

The *effective dose rate* [Sv/y] represents the annual effective dose to an average member of the critical group affected by the repository. The time evolution for the effective dose rate is depicted in FIGURE 7-1. The maximum values - evaluated over a time interval up to 10^7 years - are in the order of 10^{-9} to 10^{-5} Sv/y, i.e. the maximum values lie well below the safety limits established by the regulatory authorities within the countries of the participating organisations, ranging from 10^{-4} to $3 \cdot 10^{-4}$ Sv/y.

Whereas the maximum dose rate is similar for ENRESA-2000 and SPA-GRS, the result for Nagra's Kristallin-I calculation is lower by ca. 2 orders of magnitude, which is due to the difference in waste form and a somewhat smaller inventory. The calculations for TILA-99 yield the lowest dose rate because only releases from a single defective container are considered.

The beginning of the radiation exposure depends on the lifetime of the waste containers and the radionuclide transport times through the technical and natural barriers of the repository system. Assumed container lifetimes are 1300 years for the ENRESA-2000 study, 1000 years in the case of SPA-GRS and Kristallin-I, and 10 000 years for TILA-99 DC. Only Kristallin-I results show a significant delay of the radionuclide release (and consequently of the radiation exposure) due to the transport through the barrier system. In TILA-99 SH, an alternative release scenario with a small initial hole in a single container has been considered. In this scenario, the radiation exposure starts at very early times (< 100 years).

The maximum dose rate is determined in all cases either by fission or activation products (I-129 for ENRESA-2000 and TILA-99, C-14 in the case of SPA-GRS and Cs-135 for Kristallin-I). In all cases, the contribution of actinides to the effective dose rate is less significant and occurs only at very late release times (> 10^6 years).

7.2 Radiotoxicity concentration in biosphere water

The *radiotoxicity concentration in biosphere water* [Sv/m^3] is a measure of the radiological consequences resulting from the ingestion of water from the biosphere which is contaminated by radionuclides from the waste. Instantaneous and complete mixing of the activity released from the geosphere with the relevant water body (aquifer, surface water) in the biosphere is assumed.

The time evolution of the radiotoxicity concentrations in biosphere water is depicted in FIGURE 7-2. The maximum values of the radiotoxicity concentration resulting from the four studies are in the range of $3.7 \cdot 10^{-9}$ Sv/m^3 (TILA-99 SH) and $1.6 \cdot 10^{-6}$ Sv/m^3 (ENRESA-2000). These maximum values are well below the reference value of $2 \cdot 10^{-5}$ Sv/m^3 as adopted for the SPIN project, which comprises a broad range of different types of groundwater with naturally occurring radionuclides including deep and shallow aquifers, surface waters and water from springs.

In general, the time evolution of the curves show strong similarities with those calculated for the effective dose rates, which is an indication that the dose from the drinking water exposure path - up to a scaling factor - reflects the time-evolution of the total dose for the sum of all exposure paths.

However, the SPA-GRS study shows rather significant differences in the time-dependency of the two curves: In contrast to the effective dose rate, where the curve is determined by the dose contributions of C-14 and Cs-135 (and to a lower extent by I-129 in between these main contributions), the radiotoxicity in biosphere water is determined practically alone by the radiotoxicity of I-129 over the time range considered. This indicates, that in the SPA-GRS case other exposure paths are more important than the drinking water paths which is the only path this indicator reflects.

For the other studies, the main contribution to the radiotoxicity in biosphere water is mostly from I-129 (ENRESA-2000, TILA-99 and from Cs-135 (Kristallin-I). The radiotoxicity from actinides is rather insignificant and occurs only at very late times (> 10^6 years**).**

In TILA-99 only the drinking water exposure path is considered. This results in the fact, that the curves of the indicator differ only by a factor of 0.5 m^3/y from that of the effective dose rate.

7.3 Radiotoxicity flux from geosphere

The *radiotoxicity flux from geosphere* [Sv/y] represents a hypothetical measure of the annual radiological impact caused by ingestion of all radionuclides from the waste as they are released from the geosphere to the biosphere.

The time evolution of the radiotoxicity fluxes from the geosphere is depicted in FIGURE 7-3. The range of calculated maximum values for the radiotoxicity flux is between $3.3 \cdot 10^{-4}$ Sv/y (TILA-99 SH) and 1.6 Sv/y (both ENRESA-2000 and SPA-GRS). All the maximum values are well below the adopted reference value of 60 Sv/y for a typical surface area of 200 km^2 above a crystalline host environment.

The shape of the curves is similar to that for the radiotoxicity concentration in biosphere water; the corresponding case-specific scaling factor is the flow rate of the diluting water body in the biosphere (aquifer, surface water) for each study. The main contributions to the time-evolution of the radiotoxicity flux from geosphere stem from the same radionuclides as for the radiotoxicity concentration in biosphere water.

7.4 Time-integrated radiotoxicity flux from geosphere

The *time-integrated radiotoxicity flux from geosphere* [Sv] represents a hypothetical measure of the cumulated radiological impact to a population caused by the continuous ingestion of all radionuclides from the waste as they are released from the geosphere to the biosphere. Radioactive decay outside of the geosphere is not taken into account.

The time evolution of the time-integrated radiotoxicity fluxes from geosphere is depicted in FIGURE 7-4. Because the curves represent the time-integral of a flux, they are increasing as long as there is a release from geosphere, otherwise they are constant. The maximum value is not yet reached at the end of the calculations (10^7 years for ENRESA-2000, SPA-GRS and Kristallin-I, 10^6 years for TILA-99). At this time, the highest value is about 10^6 Sv for SPA-GRS and ENRESA-2000, 10^5 Sv for Kristallin-I, and $5 \cdot 10^2$ Sv for TILA-99 (both scenarios). As illustrated in Figure 7-4, the calculated curves are well below the reference curve according to Chapter 6, i.e. 60 Sv/y ·time.

For all studies, I-129 finally dominates the integrated flux, with the exception of Kristallin-I, where Cs-135 yields the highest contribution. In comparison, the results are rather similar for ENRESA-2000, SPA-GRS and Kristallin-I calculations; both calculations for TILA-99 DC and SH yield lower values, because only one defective container is taken into account, reducing the 'unconfined' inventory by a factor of ca. 3000.

7.5 Radiotoxicity outside geosphere

The *radiotoxicity outside geosphere* [Sv] represents a hypothetical measure of the radiological impact caused by the ingestion of all radionuclides from the waste present in the global natural environment (i.e. outside the geosphere) at any given point in time. The difference to the preceding safety indicator is that here radioactive decay/ingrowth is taken into account.

The time evolution of the radiotoxicities outside geosphere is depicted in FIGURE 7-5. As for the time-integrated radiotoxicity flux from geosphere, the calculated curves are well below the reference curve according to Chapter 6, i.e. 60 Sv/y·time.

In the examined studies, the curves for the radiotoxicity outside geosphere look very similar to those of the time-integrated radiotoxicity flux from geosphere; only at times > 10^6 years there are minor differences due to the decay of the long-lived radionuclides I-129 and Cs-135, causing a slight decrease for the SPA-GRS and the Kristallin-I study.

7.6 Relative activity concentration in biosphere water

This safety indicator was intended to introduce the time-evolution of the *relative activity concentration in biosphere water* [-]. Because nuclide-specific values for concentrations in natural waters of all safety-relevant radionuclides present in the waste are not available, relative activity concentrations in biosphere water could not be calculated. This safety indicator is therefore not further discussed in this chapter.

7.7 Relative activity flux from geosphere

The *relative activity flux from geosphere* [-] is a comparison of the time-dependent activity flux from the geosphere to some regulatory release rate constraints. In the present study the Finnish constraints according to Guide YLV 8.4 [7] were used. These values depend on the specific conditions at the proposed disposal site in Finland and the safety limit established in Finnish legislation.

The time evolution of the relative activity fluxes from the geosphere is depicted in FIGURE 7-6. The maximum values range from 10^{-4} for the SH scenario of TILA-99 up to a value above 1 for the SPA-GRS study, which is the consequence of the Finnish constraints used in these calculations; all other calculations yield peak values just below (ENRESA-2000, Kristallin-I) or well below (TILA-99) the reference value of 1 (unity).

In the case studies SPA-GRS and TILA-99, the beginning of the activity fluxes from the geosphere is directly coupled to the lifetime of the waste container. Enresa's result shows a delay of ca. 1000 years, and calculations for Kristallin-I show a significant delay of more than 10000 years of the activity release due to retardation in the barrier system.

The maximum of the calculated curves is determined in all cases either by fission products (I-129 or Cs-135) or activation products (C-14); the contributions of actinides again are rather insignificant and occur - if at all - only at very late times (> 10^6 years).

The dominating radionuclide in the ENRESA-2000 study is I-129. C-14 and Cs-135 are the main contributors in the case of SPA-GRS and TILA-99 SH. For TILA-99 DC the contributions of C-14, Sn-126, I-129 and Cs-135 are the highest. The result for vitrified high-level waste as considered in Kristallin-I is dominated by Cs-135 for the peak value and Se-79.

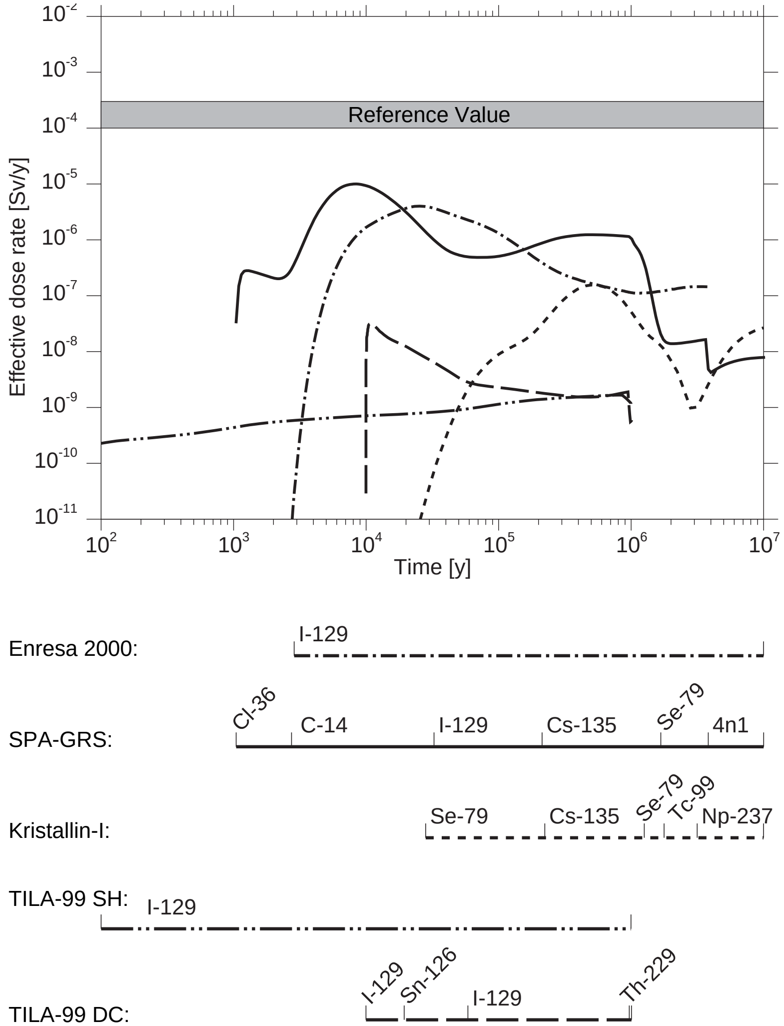

FIGURE 7-1 Effective dose rate

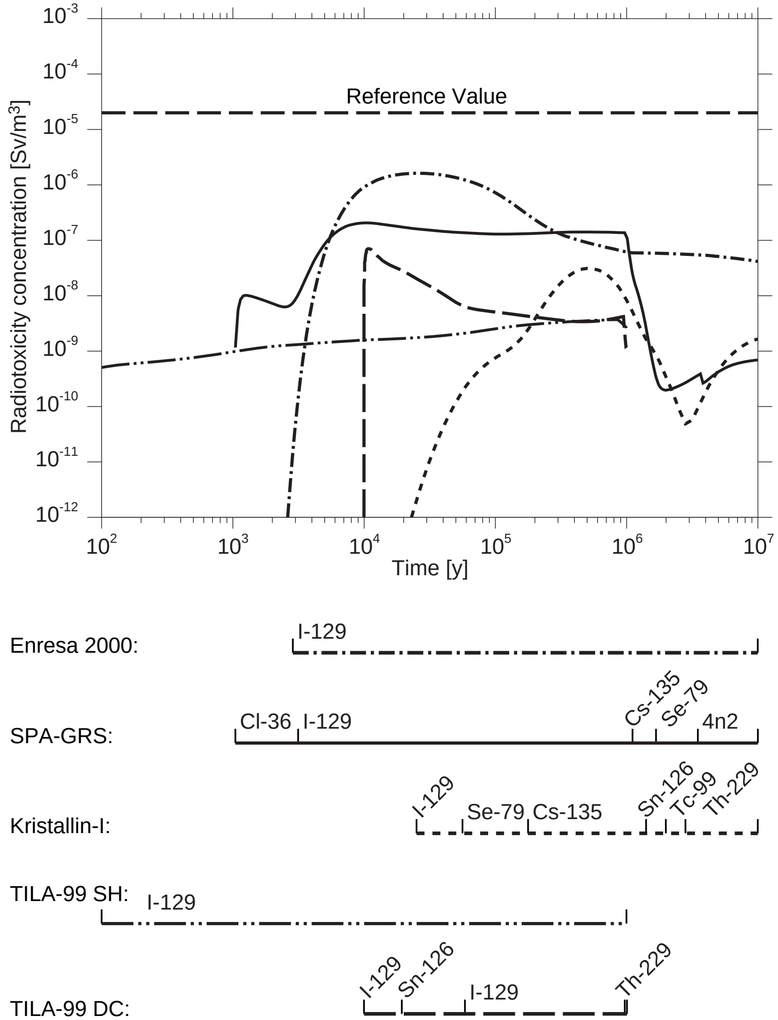

FIGURE 7-2 Radiotoxicity concentration in biosphere water

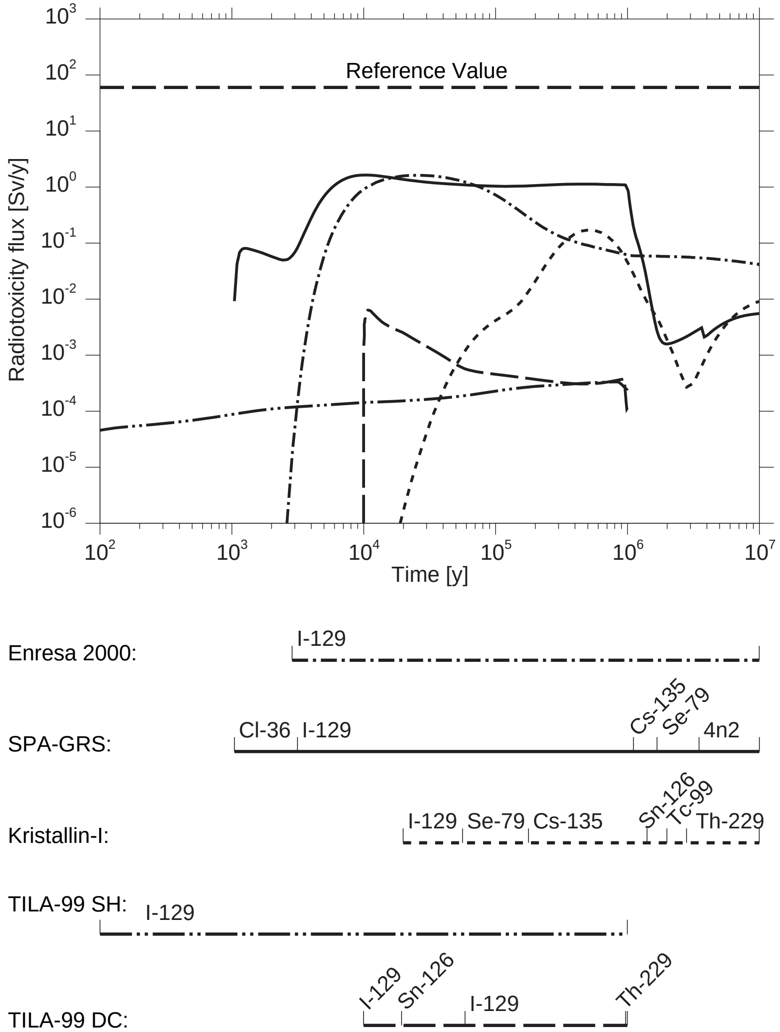

FIGURE 7-3 Radiotoxicity flux from geosphere

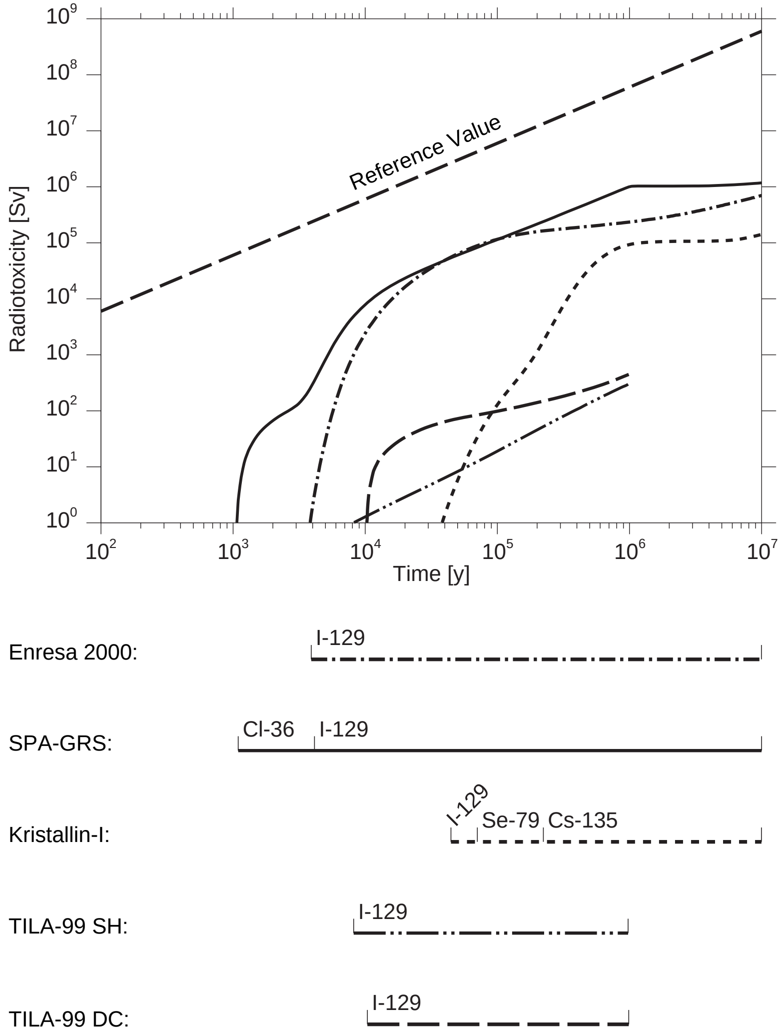

FIGURE 7-4 Time-integrated radiotoxicity flux from geosphere

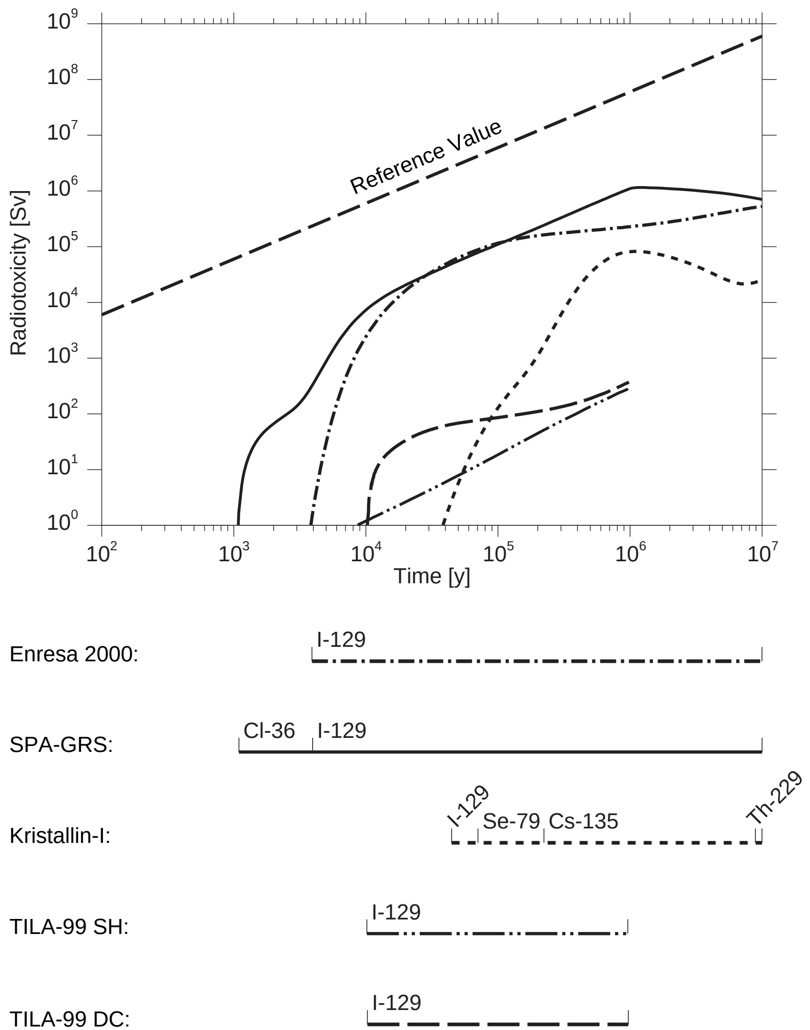

FIGURE 7-5 Radiotoxicity outside geosphere

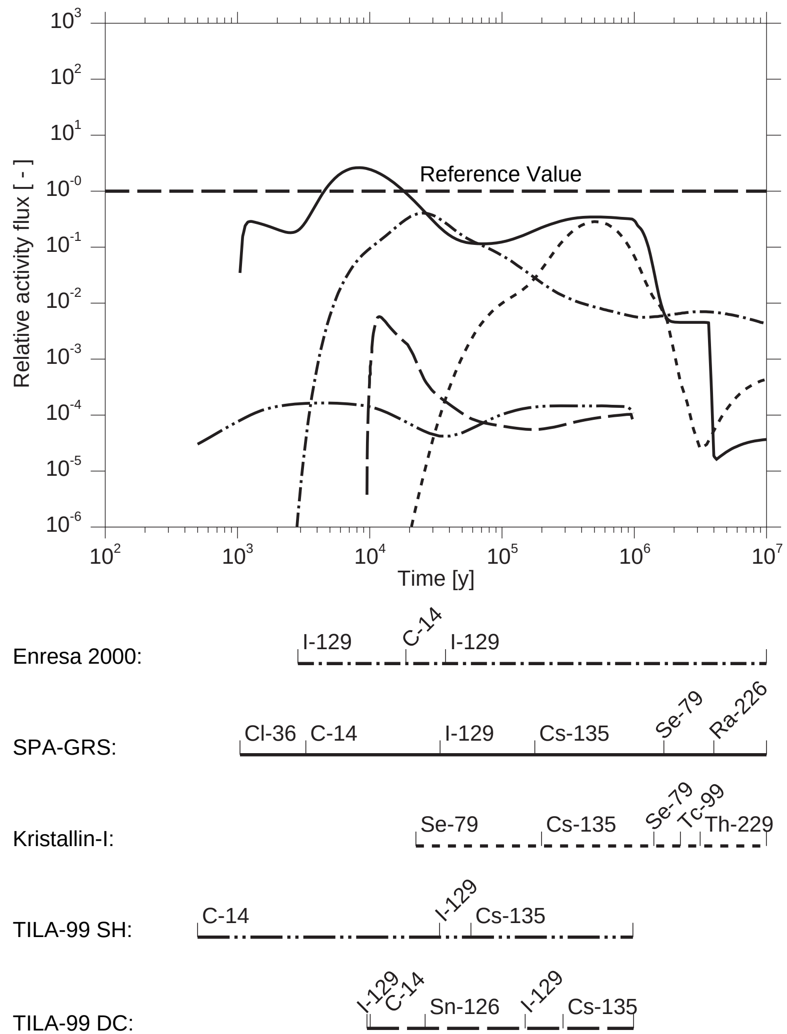

FIGURE 7-6 Relative activity flux from geosphere

8 DISCUSSION OF PERFORMANCE INDICATORS RESULTS

In this chapter, the results obtained for the performance indicators described in Chapter 3 for the four studies under consideration, are presented and discussed. Some of those indicators, however, have not been calculated and are therefore not discussed here.

To limit the number of figures as well as curves in each figure, from TILA-99 results only those dealing with the DC scenario, assuming the container disappearing after 10 000 years, are discussed in the present chapter.

Results are shown in FIGURE 8-1 to FIGURE 8-13, and explained in the following sections. In order to keep the number of figures reasonable, nuclide-specific curves are represented only for one nuclide per study. The radionuclides Sn-126 ($T_{1/2}$=1.00·10^5 y), Tc-99 ($T_{1/2}$=2.13·10^5 y), Cs-135 ($T_{1/2}$=2.30·10^6 y), I-129 ($T_{1/2}$=1.57·10^7 y) and U-238 ($T_{1/2}$=4.47·10^9 y) have been chosen to cover wide ranges of solubility, sorption and half-life. Caesium and Iodine have a high solubility, while Technetium, Tin and Uranium have low solubility. Iodine is weakly sorbed and the others have higher Kd-values.

Figures representing activities show the total activity of each nuclide, including all short-lived daughters in secular equilibrium. The activity factors given in table 5-6 have been taken into account here.

8.1 Activity in compartments

Total α- and β/γ-activities take similar values and have a similar time evolution in waste form, precipitate and buffer.

Due to their strong sorption on both engineered and natural barriers, actinides are transported through the buffer and the geosphere very slowly. Some fission and activation products show little or no sorption on repository materials, and can cross the buffer and the geosphere much faster than actinides. As a consequence, the build-up of β/γ-emitters in the geosphere and the biosphere starts much earlier than that of α-emitters.

α- activity in the geosphere increases slowly compared with β/γ-activity. Ultimately, α-activity in the geosphere reaches relatively high values, which are controlled by the Ra-226 (for spent fuel) or Th-229 (for vitrified waste) that cross the buffer. Due to their relatively short lives, these radionuclides decay in the first few meters of host formation. As a consequence, α-emitters are not uniformly distributed across the formation, but are concentrated in the host rock close to the disposal system.

Some fission and activation products have little or no sorption on granite and cross the geosphere quite quickly. Since they are β/γ-emitters, β/γ-activity in the biosphere builds up relatively quickly, although never reaches very high values. Since long-lived α-emitters have strong sorption on granite, it takes a long time before α-activity in the biosphere reaches a significant value, and this value is always quite low.

Since both are highly soluble radionuclides, the masses of I-129 and Cs-135 in the precipitate and surrounding water inside the waste package are negligible. Sorption of Cs-135 onto the buffer and the geosphere materials results in significant amounts of Cs-135 in both compartments. The zero or very low Kd-values for I-129 in the buffer and the geosphere cause the amounts of I-129 in these compartments to be extremely low. At any instant, the I-129 inventory in the system is either immobilised in the waste form or is in the biosphere. For this reason doses from spent fuel repositories are controlled by I-129.

For low solubility radionuclides, such as Sn-126 and U-238, nearly all the inventory is immobilised in the waste form or precipitated inside the waste package. Due to the large amount of uranium in the waste form, U-238 is always controlled by its low solubility, and only a small fraction of the inventory has started to move after 10^7 years. Sn-126 transport is solubility controlled during the first two hundred thousand years, and after that instant Sn-126 in precipitate decreases rapidly.

Due to strong sorption on buffer and geosphere materials, a large proportion of the Sn-126 and U-238 that start to move are sorbed onto the buffer or the geosphere, and only a very small fraction has reached the biosphere up to 10^7 years. For relatively short-lived radionuclides, such as Sn-126, precipitation and strong sorption delay the transport long enough to allow most of the inventory to decay before reaching the biosphere. For very long-lived radionuclides, such as U-238, little decay can be expected. After 10^7 years, U-238 transport has reached a steady-state, activities in the buffer and the geosphere are constant, and U-238 activity flux from geosphere is constant too (see Figure 8-5). Inevitably, significant amounts of U-238 will finally reach the biosphere after a very long time period.

8.2 Activity flux from compartments

In the three sets of calculations with spent fuel (SPA-GRS, ENRESA-2000 and TILA-99) total activity fluxes from the waste form and the waste package show initial spikes after container failure. These spikes are caused by the Instant Release Fraction (IRF) of the inventory, formed by the radionuclides in the gap and the grain boundaries of spent fuel. No spikes are seen in Kristallin-I, since no IRF is assumed for vitrified wastes.

In ENRESA-2000 and TILA-99 the initial spikes in the fluxes from the waste form and the waste package are caused only by β/γ-emitters. In SPA-GRS both β/γ- and α-emitters show similar spikes. The differences are caused by the different IRF´s used: SPA-GRS includes some actinides while ENRESA-2000 and TILA-99 only consider fission and activation products.

Transport through the buffer spreads any instantaneous injection over a long time period. As a consequence, fluxes from the near field and geosphere do not show an initial spike.

α- and β/γ-activity fluxes from the waste form and the waste package reach the peak values immediately after container failure. β/γ-activity fluxes from buffer reach high values quite fast, while releases from buffer of α-emitters start much later.

The three partners dealing with spent fuel have obtained the same result: long term releases from buffer are controlled by Ra-226. Since Ra-226 chain has 5 α- and 4 β/γ-decays, for spent fuel roughly the same α- and β/γ-activity fluxes cross the buffer in the long term.

Due to its relatively short half-life ($T_{1/2}$=1600 years) and significant sorption on granite in non-saline groundwater, Ra-226, which has the highest flux into the geosphere, decays during its transit through the geosphere. Long lived actinides, which are α-emitters, cross the geosphere after a long time due to their strong sorption on granite.

Fluxes from the geosphere are controlled by weakly sorbed or non-sorbed fission and activation products, which are β/γ-emitters, during the first few million years. Only after this time, releases of α-emitters become significant.

Total β/γ-activity fluxes from the geosphere are far greater (three or more orders of magnitude of difference in peak values) and start much earlier than total α-activity fluxes.

In all cases the geosphere is a very effective barrier to delay and limit the releases of α-emitters to the biosphere: α-activity fluxes from geosphere are negligible up to 10^5 years and peak α-activity fluxes are always very low ($<10^5$ Bq/y).

Highly soluble and weakly sorbed species, such as I-129, cross the barrier system quite quickly. As a consequence, after a brief transient period, releases from the geosphere are controlled by releases from the waste. Since radioactive decay is negligible, the release rates from the consequent barriers become equal to the release rate from the waste.

Highly soluble and sorbed species, such as Cs-135, start to move immediately after release from the waste form, but need long time periods to cross the barrier system. This allows for significant radioactive decay during the transport if the radionuclide half-life is short enough, but this is not the case for Cs-135. Although Cs-135 peak release rates from the buffer and geosphere are similar to peak release rate from the waste form, they are significantly delayed.

Low solubility and strongly sorbed species, such as U-238, need much time to cross the barrier system. Due to the low solubility, fluxes from the barriers are independent of the releases from the waste. After a long transient period, a steady state is reached in which all the fluxes entering and leaving each barrier are similar, because radioactive decay is negligible.

In SPA-GRS the U-238 flux from the waste package becomes negative between $4 \cdot 10^5$ and $2 \cdot 10^6$ years. This is because no solubility limits are considered in the buffer. Therefore, U-238 concentration in buffer water can, caused by decay of Pu-242, increase above the solubility limit in the waste package, which results in a negative flux.

Sn-126 is solubility controlled during a part of the considered time period. The Sn-126 release rate from the waste form is greater than the release rate form the waste package up to 50,000 years and a fraction of the released Sn-126 precipitates. During the period from 50 000 and 200 000 years, precipitated Sn-126 dissolves and transport remains controlled by solubility. After 200 000 years, Sn-126 behaves as a non-solubility controlled species: releases from the waste form and the waste package are equal.

8.3 Time-integrated activity flux from compartments

In the four sets of calculations the time-integrated total α- and β/γ-activity fluxes from compartments behave in a similar way to the time-integrated radiotoxicity fluxes from compartments. The reason is that short-lived daughters of Ra-226 and Th-229 are both α- and β/γ-emitters. Only the time-integrated total α-activity flux from the geosphere behaves differently to the time-integrated radiotoxicity flux, because all long-lived α-emitters are strongly retarded by the geosphere.

For individual radionuclides not included in a decay chain, the time-integrated activity fluxes from compartments graphically show the amount of the radionuclide that has crossed each barrier at each instant. When the time-integrated flux becomes constant after some time, the total activity that crosses the barrier is obtained. Comparing the time-integrated fluxes that enter and leave a barrier, the amount of radioactive decay in the barrier can be estimated.

For highly soluble radionuclides, such as I-129 and Cs-135, the masses released from the waste immediately enter the buffer, and time-integrated fluxes from the waste form and the waste package are equal. Figure 8-6 clearly shows the different behaviour of sorbed (Cs-135) and non-sorbed species (I-129) not controlled by solubility limits. I-129 rapidly crosses the buffer and the geosphere and all the inventory released from the waste reaches the biosphere. Sorbed species move very slowly and can suffer a significant decay during the transport through the barriers. Cs-135, however, is too long-lived to show this effect.

For low solubility radionuclides, such as Sn-126 and U-238, curves for time-integrated fluxes from the waste form and the waste package are different. Since the mass of U-238 in the spent fuel is about ten thousand times the mass of Sn-126, low solubility is a far more important barrier for U-238 than for Sn-126. While nearly all the Sn-126 released from the waste finally enters the buffer, after 10^7 years only 0.1% of the Uranium in the repository has started to mobilize.

Although it is logical to expect that time-integrated fluxes can only increase with time, there is a small decrease of time-integrated U-238 flux from waste package in SPA-GRS after $4 \cdot 10^{-5}$ years. This is due to the effect described in Section 8.3.

8.4 Radiotoxicity in compartments

The time evolution of total radiotoxicity is quite similar for the three partners dealing with spent fuel (ENRESA-2000, SPA-GRS and TILA-99). Since the radionuclide inventories per ton of heavy metal are quite similar (see Table 5-6), differences in total radiotoxicity are caused only by the different amounts of heavy metal considered in the calculations. In SPA-GRS and ENRESA-2000 the time evolution of total radiotoxicity are very similar, because SPA-GRS calculations are done for 6250 t_{HM} (1/4 of the 25000 t_{HM} disposed in the repository) and ENRESA-2000 calculations are performed for 6640 t_{HM}. TILA-99 calculations are done for a single container with 2.14 t_{HM}, and as a consequence, at any instant there exists a factor 3000 of difference with SPA-GRS and ENRESA-2000 total radiotoxicities. From 100 to 10^7 years, spent fuel radiotoxicity decreases by more than three orders of magnitude.

Total radiotoxicity in Kristallin-I has a different behaviour because the waste is vitrified HLW that contains only a fraction of the radiotoxicity in the spent fuel. Kristallin-I repository contains the vitrified wastes arising from the reprocessing of 3730 t_{HM} (more than half of the amount of heavy metal considered by ENRESA-2000 and SPA-GRS), and total radiotoxicity is always less than 10% of total radiotoxicity in ENRESA-2000 and SPA-GRS cases. From 100 to 10^7 years, vitrified HLW radiotoxicity decreases by nearly four orders of magnitude.

In the four sets of calculations most of the radiotoxicity remains immobilized in the waste form up to 10^5 years. In three cases (Kristallin-I, SPA-GRS and TILA-99) after 10^6 years 100% of the waste form is assumed to be dissolved and no radiotoxicity remains immobilized in the waste. With more realistic models, a significant amount of total radiotoxicity would remain immobilized in the waste up to 10^7 years and longer (ENRESA-2000 results).

In all the calculations most of the radiotoxicity released from the waste remains in the near field (precipitated or in the buffer) up to 10^7 years.

In ENRESA-2000 and SPA-GRS, radiotoxicity in the precipitate is greater than radiotoxicity in the buffer up to 10^5 years. After 10^5 years, in SPA-GRS radiotoxicity in the precipitate and the buffer are similar, while in ENRESA-2000 radiotoxicity in the precipitate is ten times smaller than in the buffer. In Kristallin-I and TILA-99, radiotoxicity in the precipitate is roughly one order of magnitude smaller than in the buffer at any instant.

In spent fuel repositories, radiotoxicity in the precipitate is controlled by plutonium isotopes up to 10^5 years and by Np-237 and daughters afterwards. Radiotoxicity in the buffer is controlled by Am-241 during several thousands of years, plutonium isotopes up to 10^5 years and Ra-226 afterwards.

In all cases, the radiotoxicities in the geosphere and the biosphere represent a small fraction of total radiotoxicity at any instant.

Radiotoxicities in the geosphere are controlled by Ra-226 ($T_{1/2}$=1600 years) and Th-229 ($T_{1/2}$=7344 years) created by radioactive decay in the precipitate that manage to cross the buffer in important amounts. Due to their relatively short lives, these radionuclides will decay in the first few metres of host formation.

Maximum value of total radiotoxicity in the biosphere is between 10^5 and 10^6 Sv in SPA-GRS, Kristallin-I and ENRESA-2000. Since TILA-99 only considers one container with 2.14 t_{HM}, there are three orders of magnitude of difference compared with the results of the other partners. Radiotoxicity in biosphere is controlled by non-sorbed or weakly sorbed fission products: I-129 in spent fuel repositories and Cs-135 in vitrified HLW repositories.

8.5 Radiotoxicity flux from compartments

This indicator shows that the consecutive barriers in a repository gradually reduce the radiotoxicity fluxes. Radiotoxicity flux from one barrier is always smaller than the flux from the preceding barrier, with just one exception: releases from the waste form become smaller than the releases from the waste package after some time.

In the four sets of calculations, radiotoxicity fluxes from the waste package take very high values when the container fails, then slowly decrease and after approximately 10^5 years increase again. This final increasing phase is caused by Ra-226 ($T_{1/2}$=1600 years) and Th-229 ($T_{1/2}$=7344 years) created in the precipitate from decay of U-234/Th-230 and Np-237/U-233, that quickly diffuse into the buffer.

Ra-226 and Th-229 decay during the transport through the buffer, but a significant fraction crosses the barrier and again these two radionuclides (mainly Ra-226) control the radiotoxicity flux leaving the buffer.

The geosphere is a very effective barrier to movement of Ra-226 and Th-229. As a consequence, radiotoxicity fluxes from geosphere are controlled by weakly sorbed fission and activation products, and only in the long term the contribution of actinides and their daughters becomes significant.

In all calculations the following results have been obtained:

- Long-term radiotoxicity fluxes from the buffer are one or two orders of magnitude smaller than long-term radiotoxicity fluxes from the waste package.
- Long-term radiotoxicity fluxes from the geosphere are three to five orders of magnitude (spent fuel) or two orders of magnitude (vitrified waste) smaller than long term radiotoxicity fluxes from the buffer.
- Peak radiotoxicity fluxes from the geosphere are very low in all cases: 0.1 to 1 Sv/yr.

8.6 Time-integrated radiotoxicity flux from compartments

In the four sets of calculations, long-term time-integrated radiotoxicity fluxes from the waste package and buffer can be more than two orders of magnitude greater than the radiotoxicity outside the same compartment. Long-term differences between these two indicators for the waste form can reach a factor 10. Differences between them for the geosphere are small because releases to biosphere are controlled by long lived fission products, such as I-129 and Cs-135.

The time-integrated radiotoxicity flux does not take into account the decay of radionuclides after crossing the interface between barriers. For this reason the time-integrated radiotoxicity flux from a compartment can reach much higher values than the equivalent radiotoxicity outside the compartment. In fact, if calculations continue beyond 10^7 years, time-integrated radiotoxicity fluxes out of the waste package can reach values greater than the radiotoxicity in the waste after 100 years of cooling.

The largest contributions to the time-integrated radiotoxicity fluxes from the waste package are caused by relatively short-lived radionuclides (Ra-226 or Th-229) created inside the waste package by radioactive decay of their precipitated parents (U-234/Th-230 and Np-237/U-233). These daughters quickly diffuse into the buffer. Significant amounts of Ra-226 and Th-229 cross the buffer and control the time-integrated radiotoxicity fluxes from the buffer.

The above effect can be better understood with the following example. Assuming a constant activity flux (AF) from a compartment for a single radionuclide, the activity outside the compartment will increase until reaching a peak value equal to $AF \cdot T_{1/2}/\ln(2)$. The time-integrated activity flux during a time period T would be simply $AF \cdot T$. Applying the above expressions to Ra-226 ($T_{1/2}$=1600 years), it is found that after 1 million years the time-integrated activity flux would be 400 times the activity outside the compartment.

8.7 Radiotoxicity outside compartments

This indicator provides similar information to radiotoxicity in compartments. It is an alternative form of presenting results, that can be useful to graphically show the period of time during which a given barrier is relevant.

In SPA-GRS, TILA-99 and Kristallin-I total radiotoxicity and radiotoxicity outside the waste form are equal after 10^6 years. The waste form plays no role as a barrier after 10^6 years for these three partners, as the waste form is assumed to be totally degraded after that time. In ENRESA-2000 the waste form remains a useful barrier up to 10^7 years.

In Kristallin-I and TILA-99 radiotoxicities outside the waste form and outside the waste package are roughly the same at any instant. In ENRESA-2000 both magnitudes are equal after $2 \cdot 10^5$ years. In GRS calculations, difference is smaller than a factor 2 after 10^5 years. These results can suggest that low solubilities are not important in limiting nuclide transport, but this would be a wrong conclusion. Uranium isotopes are released from the waste precipitate and produce short-lived daughters that quickly pass towards the buffer. Since the ingestion dose coefficients of these daughters are much greater than those of the parents, most of the radiotoxicity will be in the buffer. If no solubility limit for uranium is imposed, radiotoxicity outside the waste package would be slightly greater but radiotoxicity outside the buffer and the geosphere could be very much greater.

In the four sets of calculations, radiotoxicities outside the buffer are much smaller than radiotoxicities outside the waste package at any instant. This result shows the efficiency and importance of the buffer barrier up to 10^7 years.

Radiotoxicities outside the geosphere increase with time, but even after 10^7 years only a small fraction (0.1 to 1%) of remaining radiotoxicity has reached the biosphere. Due to the models of reversible sorption onto buffer and granite adopted, if the calculations continue long enough, 100% of total radiotoxicity will finally reach the biosphere, though only after a very long time.

8.8 Activity concentrations in biosphere water and waste package water

In all calculations, for any radionuclide the concentration in waste package water is at least six or seven orders of magnitude greater than the concentration in biosphere water. These differences can be explained in general terms as controlled by the flow rates in the repository near field and the biosphere.

For a stable solute, if there is a constant concentration C_0 in the inner surface of buffer and a small water flow Q_{NF} in the near field (mixing tank boundary condition), when steady-state conditions are reached, the concentration gradient through the buffer is small. Solute concentration at the outer surface of buffer becomes roughly equal to C_o and the mass flux leaving the buffer is the product $Q_{NF} \cdot C_0$.

In steady-state conditions, mass flux leaving the buffer will be equal to the mass flux entering the geosphere, that is the product of the concentration in biosphere water (C_{BIO}) times the dilution flow in the biosphere (Q_{BIO}). As a consequence the ratio between the concentrations in biosphere and waste package water is equal to the quotient Q_{BIO}/Q_{NF}, that can be considered as a dilution factor. The value of this magnitude for each study is the 'Dilution factor from near field to biosphere' in TABLE 5-5.

For transient conditions, radioactive solutes or high solubility species, the dilution factor will be greater than Q_{BIO}/Q_{NF}. As a consequence, this magnitude is a good estimation of the minimum dilution factor from the container to the biosphere.

The instant release inventory in spent fuel includes several high solubility radionuclides (I-129, Cl-36, Cs-135). Therefore, after container failure, the concentrations of these radionuclides in waste package water can reach very high values during a short time period, until they diffuse into the buffer. Transport through the barrier system flattens the initial spikes of concentration in waste package water, and no spikes are seen in the concentrations in biosphere water.

8.9 Transport times through compartments

The radionuclide transport time quantifies the capability of the barrier to delay the releases of the radionuclide. Comparison of radionuclide half-live with its transport time through a barrier provides information about the amount of decay that the radionuclide will suffer during its transit through the barrier. If the transport time is much greater than the radionuclide half-life (10 times or more), only a very small fraction of the radionuclide mass that enters the barrier finally crosses it.

Transport times through the buffer and the biosphere have been calculated, and the results are shown in TABLE 8-1 and FIGURE 8-13.

Transport time through the buffer is proportional to the square of the thickness of the buffer, the retardation factor and the inverse of the pore diffusion coefficient. Transport time through the geosphere is proportional to the water travel time and the retardation factor, assuming that diffusion into the matrix is relatively fast compared with advection in the fractures.

The retardation factor is 1 for non-sorbed species, and for sorbed species is roughly proportional to the distribution coefficient (Kd) between the solid (buffer or granite) and water.

For most radionuclides, transport time through the geosphere is greater than transport time through the buffer. Since actinides and daughters are strongly sorbed in buffer and granite, in general their transport times are longer than those calculated for fission and activation products.

TABLE 8-1 Transport times [y] through buffer and geosphere (alphabetical order)

	$T_{1/2}$ [y]	ENRESA-2000		SPA-GRS		Kristallin-I		TILA-99 DC	
		Buffer	Geosphere	Buffer	Geosphere	Buffer	Geosphere	Buffer	Geosphere
Am-241	$4.32\cdot10^{2}$	$1.25\cdot10^{6}$	$1.51\cdot10^{7}$	$7.90\cdot10^{4}$	$2.35\cdot10^{7}$	$1.26\cdot10^{5}$	$8.91\cdot10^{6}$	$1.29\cdot10^{5}$	$5.19\cdot10^{5}$
Am-242m	$1.52\cdot10^{2}$								
Am-243	$7.38\cdot10^{3}$								
C-14	$5.73\cdot10^{3}$	$1.00\cdot10^{2}$	$3.90\cdot10^{4}$	$1.60\cdot10^{2}$	$4.75\cdot10^{3}$	--	--	$1.25\cdot10^{2}$	$4.71\cdot10^{2}$
Ca-41	$8.10\cdot10^{4}$	$3.60\cdot10^{3}$	$3.12\cdot10^{5}$	$3.25\cdot10^{3}$	$4.60\cdot10^{4}$	--	--	--	--
Cl-36	$3.01\cdot10^{5}$	$7.40\cdot10^{1}$	$8.40\cdot10^{3}$	$3.00\cdot10^{0}$	$1.10\cdot10^{2}$	--	--	$1.25\cdot10^{2}$	$3.11\cdot10^{1}$
Cm-244	$1.81\cdot10^{1}$	$1.25\cdot10^{6}$	$1.50\cdot10^{7}$	$7.80\cdot10^{4}$	$2.35\cdot10^{7}$	$1.26\cdot10^{5}$	$8.91\cdot10^{6}$	$1.29\cdot10^{5}$	$5.19\cdot10^{5}$
Cm-245	$8.50\cdot10^{3}$								
Cm-246	$4.73\cdot10^{3}$								
Cm-247	$1.56\cdot10^{7}$								
Cm-248	$3.39\cdot10^{5}$								
Cs-135	$2.30\cdot10^{6}$	$3.40\cdot10^{4}$	$3.04\cdot10^{6}$	$1.60\cdot10^{2}$	$1.98\cdot10^{5}$	$2.24\cdot10^{2}$	$3.98\cdot10^{5}$	$2.19\cdot10^{4}$	$6.39\cdot10^{5}$
Cs-137	$3.00\cdot10^{1}$								
Ho-166m	$1.20\cdot10^{3}$	$3.38\cdot10^{5}$	$1.50\cdot10^{7}$	--	--	--	--	--	--
I-129	$1.57\cdot10^{7}$	$7.30\cdot10^{1}$	$8.40\cdot10^{3}$	$7.50\cdot10^{1}$	$4.80\cdot10^{3}$	$3.16\cdot10^{2}$	$7.08\cdot10^{4}$	$1.25\cdot10^{2}$	$3.11\cdot10^{1}$
Mo-93	$3.50\cdot10^{3}$	$7.30\cdot10^{3}$	$4.00\cdot10^{4}$	$7.50\cdot10^{1}$	$4.70\cdot10^{4}$	--	--	--	--
Nb-94	$2.03\cdot10^{4}$	$8.30\cdot10^{4}$	$1.50\cdot10^{7}$	$1.60\cdot10^{4}$	$4.70\cdot10^{4}$	--	--	$2.19\cdot10^{4}$	$2.79\cdot10^{5}$
Ni-59	$8.00\cdot10^{4}$	$1.85\cdot10^{4}$	$6.04\cdot10^{6}$	$1.60\cdot10^{4}$	$2.30\cdot10^{6}$	$1.78\cdot10^{4}$	$4.47\cdot10^{6}$	$9.33\cdot10^{4}$	$1.24\cdot10^{6}$
Ni-63	$9.20\cdot10^{1}$								
Np-236	$1.54\cdot10^{5}$	$5.08\cdot10^{5}$	$1.50\cdot10^{7}$	$7.80\cdot10^{4}$	$4.70\cdot10^{6}$	$1.26\cdot10^{5}$	$7.08\cdot10^{6}$	$1.10\cdot10^{5}$	$2.44\cdot10^{6}$
Np-237	$2.14\cdot10^{6}$								
Pa-231	$3.28\cdot10^{4}$	$6.90\cdot10^{4}$	$6.04\cdot10^{6}$	$1.60\cdot10^{4}$	$4.70\cdot10^{6}$	$2.51\cdot10^{4}$	$7.08\cdot10^{6}$	$9.33\cdot10^{4}$	$6.39\cdot10^{5}$
Pd-107	$6.50\cdot10^{6}$	$4.30\cdot10^{4}$	$3.12\cdot10^{5}$	$1.60\cdot10^{4}$	$2.30\cdot10^{6}$	$2.51\cdot10^{4}$	$4.47\cdot10^{6}$	$1.95\cdot10^{3}$	$4.25\cdot10^{3}$
Pu-238	$8.78\cdot10^{1}$	$1.40\cdot10^{6}$	$3.01\cdot10^{7}$	$7.90\cdot10^{4}$	$2.35\cdot10^{7}$	$1.26\cdot10^{5}$	$1.00\cdot10^{7}$	$1.29\cdot10^{5}$	$6.00\cdot10^{6}$
Pu-239	$2.41\cdot10^{4}$								
Pu-240	$6.54\cdot10^{3}$								
Pu-241	$1.44\cdot10^{1}$								
Pu-242	$3.87\cdot10^{5}$								
Pu-244	$8.26\cdot10^{7}$								
Ra-226	$1.60\cdot10^{3}$	$6.10\cdot10^{3}$	$9.04\cdot10^{6}$	$1.60\cdot10^{2}$	$2.35\cdot10^{6}$	$2.24\cdot10^{2}$	$4.47\cdot10^{6}$	$1.15\cdot10^{4}$	$2.44\cdot10^{6}$
Rb-87	$4.70\cdot10^{10}$	$4.40\cdot10^{3}$	$7.03\cdot10^{4}$	$1.60\cdot10^{2}$	$1.98\cdot10^{5}$	--	--	--	--
Se-79	$1.10\cdot10^{6}$	$1.50\cdot10^{3}$	$3.90\cdot10^{4}$	$8.00\cdot10^{1}$	$4.70\cdot10^{4}$	$5.01\cdot10^{3}$	$8.91\cdot10^{4}$	$1.25\cdot10^{2}$	$2.04\cdot10^{3}$
Sm-147	$1.07\cdot10^{11}$	$3.40\cdot10^{5}$	$1.50\cdot10^{7}$	$7.80\cdot10^{4}$	$2.30\cdot10^{7}$	--	--	$9.33\cdot10^{4}$	$2.79\cdot10^{5}$
Sm-151	$9.00\cdot10^{1}$								
Sn-126	$1.00\cdot10^{5}$	$3.30\cdot10^{5}$	$3.10\cdot10^{5}$	$1.60\cdot10^{4}$	$2.30\cdot10^{6}$	$2.51\cdot10^{4}$	$8.91\cdot10^{5}$	$1.95\cdot10^{3}$	$4.25\cdot10^{3}$
Sr-90	$2.91\cdot10^{1}$	$3.60\cdot10^{3}$	$3.10\cdot10^{5}$	$1.70\cdot10^{2}$	$4.70\cdot10^{4}$	--	--	$6.02\cdot10^{3}$	$6.55\cdot10^{4}$
Tc-99	$2.13\cdot10^{5}$	$5.80\cdot10^{4}$	$3.04\cdot10^{6}$	$2.50\cdot10^{3}$	$2.30\cdot10^{6}$	$2.51\cdot10^{3}$	$8.91\cdot10^{5}$	$1.86\cdot10^{4}$	$6.39\cdot10^{5}$
Th-229	$7.34\cdot10^{3}$	$1.25\cdot10^{6}$	$1.50\cdot10^{7}$	$7.60\cdot10^{4}$	$4.70\cdot10^{6}$	$1.26\cdot10^{5}$	$7.08\cdot10^{6}$	$1.29\cdot10^{5}$	$2.44\cdot10^{6}$
Th-230	$7.70\cdot10^{4}$								
Th-232	$1.40\cdot10^{10}$								
U-233	$1.58\cdot10^{5}$	$2.95\cdot10^{5}$	$1.50\cdot10^{7}$	$7.90\cdot10^{4}$	$4.70\cdot10^{6}$	$2.51\cdot10^{4}$	$7.08\cdot10^{6}$	$9.33\cdot10^{4}$	$1.24\cdot10^{6}$
U-234	$2.44\cdot10^{5}$								
U-235	$7.04\cdot10^{8}$								
U-236	$2.34\cdot10^{7}$								
U-238	$4.47\cdot10^{9}$								
Zr-93	$1.53\cdot10^{6}$	$5.85\cdot10^{5}$	$3.01\cdot10^{7}$	$1.60\cdot10^{4}$	$4.80\cdot10^{6}$	$2.51\cdot10^{4}$	$7.08\cdot10^{6}$	$2.09\cdot10^{5}$	$2.44\cdot10^{6}$

8.10 Proportion of not totally isolated waste

In some performance assessments, it is assumed that only a part of the disposed waste contributes to the release of radionuclides. The rest is presumed to be retained in the containers or in very slow transport paths which are not taken into account. In TILA-99, radionuclide transport analyses are performed for a single defective container. The total number of copper-iron containers in the repository is approx. 1400. The calculation thus deals with a proportion of about $7 \cdot 10^{-4}$ of the total inventory of the repository. In SPA-GRS it is assumed that only 25% of the containers are affected by the water flow, the rest can be considered to be totally isolated from the geosphere. The indicator has therefore the value 0.25. For the other studies under consideration, it has a value of 1.

8.11 Time-integrated flux from geosphere divided by initial inventory

Time-integrated fluxes of activity and radiotoxicity from compartments are shown in figures 8-6, 8-7 and 8-10 . The indicator discussed here is in principle the same, calculated for the geosphere, except that the value is divided by the fraction of the initial inventory (i.e. the inventory at repository closure) which contributes to the release of contaminants, in order to yield the fraction of activity or radiotoxicity ever reaching the biosphere. The curves for short-lived nuclides reach an asymptotic value within the modelled time, those of the long-lived nuclides continue to increase. The radiotoxicity curves also continue to increase even after very long times, which is due to the continuing production of Ra-226 from the very long-lived U-238. To derive a time-independent value for the indicator, the integrated activity or radiotoxicity after 10^7 years (10^6 years in TILA-99 case) is used. These values are compiled in TABLE 8-2.

TABLE 8-2 Fractional release from geosphere

Study		Initial value	Integrated flux	Fraction
ENRESA-2000	Radiotoxicity	$7.97 \cdot 10^{11}$	$6.80 \cdot 10^{5}$	$8.5 \cdot 10^{-7}$
	Activity Sn-126	$2.32 \cdot 10^{14}$	$6.10 \cdot 10^{8}$	$6.6 \cdot 10^{-6}$
SPA-GRS	Radiotoxicity	$7.34 \cdot 10^{11}$	$1.19 \cdot 10^{6}$	$1.6 \cdot 10^{-6}$
	Activity U-238	$7.25 \cdot 10^{13}$	$5.88 \cdot 10^{9}$	$8.1 \cdot 10^{-5}$
Kristallin-I	Radiotoxicity	$1.67 \cdot 10^{10}$	$1.38 \cdot 10^{5}$	$8.3 \cdot 10^{-6}$
	Activity Cs-135	$5.40 \cdot 10^{13}$	$5.10 \cdot 10^{13}$	0.94
TILA-99 DC	Radiotoxicity	$3.08 \cdot 10^{8}$	$4.49 \cdot 10^{2}$	$1.5 \cdot 10^{-6}$
	Activity I-129	$2.78 \cdot 10^{9}$	$2.73 \cdot 10^{9}$	0.98

It can be seen that in all cases only a very small fraction of the initial radiotoxicity reaches the biosphere. The complementary fraction has decayed during the transport through the barrier system. Long-lived and fast migrating radionuclides, however, are nearly completely released to the biosphere.

8.12 Concentration in biosphere water divided by concentration in waste package water

The indicator has been calculated only for SPA-GRS, for demonstration purpose. It is presented in FIGURE 8-1, calculated for a sorbing and a non-sorbing stable nuclide. For the sorbing nuclide the distribution coefficient of Americium has been used. Both the biosphere water concentration and, for comparison, the concentration in the excavation damaged zone (EDZ), each normalised to the waste package water concentration, are presented in the figure. It can be seen that the same asymptotic values are reached for sorbing and non-sorbing elements. This happens early for the non-sorbing nuclide, and late for the sorbing one. The relative EDZ water concentrations reach a value very close to 1, because of the low groundwater flow. The relative biosphere water concentrations, however, provide a measure for the total dilution in geosphere and biosphere, which is about $4.8 \cdot 10^{-7}$, i.e. the inverse of the dilution factor given in TABLE 5-5.

Using SPA-GRS study as an example, the following chain of statements can be derived, which may be useful for public communication:

- only 25% of the disposed contaminants has the potential to reach the biosphere,
- of this amount only a small proportion of $1.6 \cdot 10^{-6}$ of the radiotoxicity will reach the biosphere,
- nuclide concentration is diluted to a fraction of less than $5 \cdot 10^{-7}$ from the waste package to the biosphere.

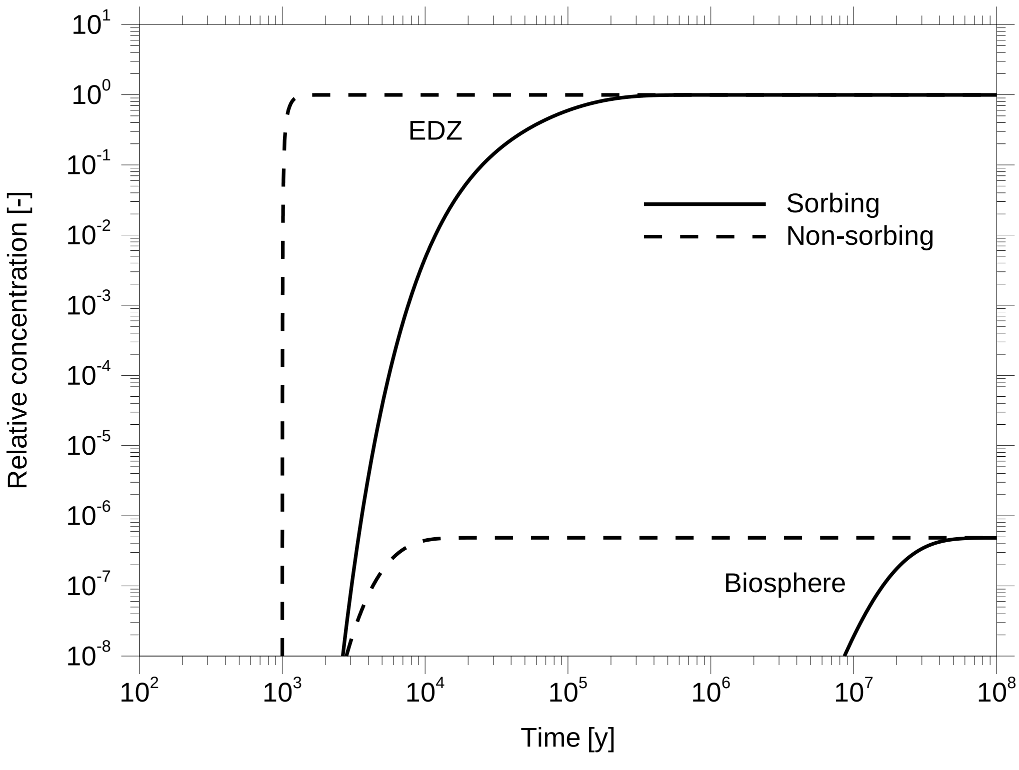

FIGURE 8-1 Relative concentrations in the EDZ and the biosphere, calculated for a sorbing and a non-sorbing nuclide

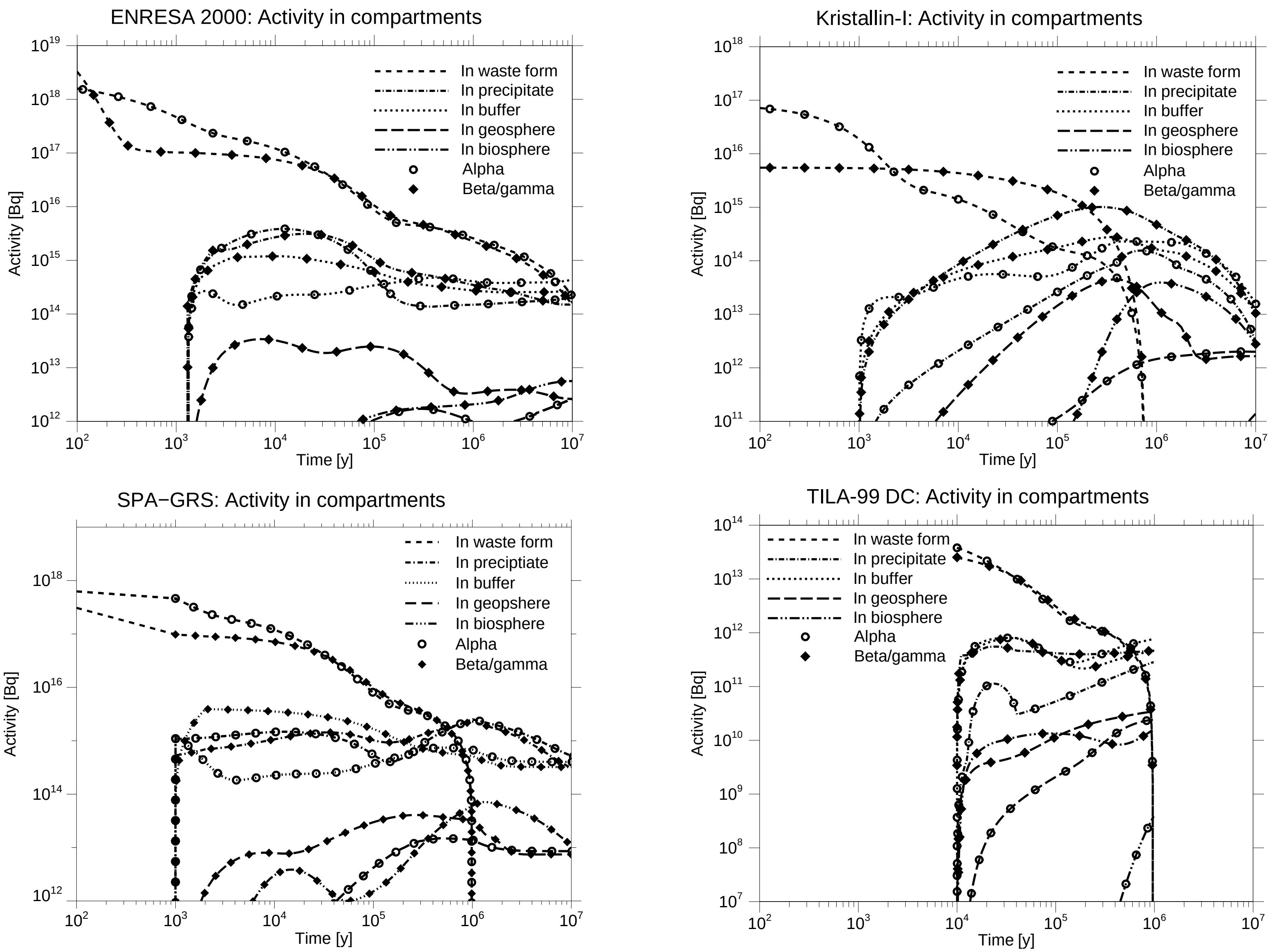

FIGURE 8-2 Total α- and β/γ-activities in compartments

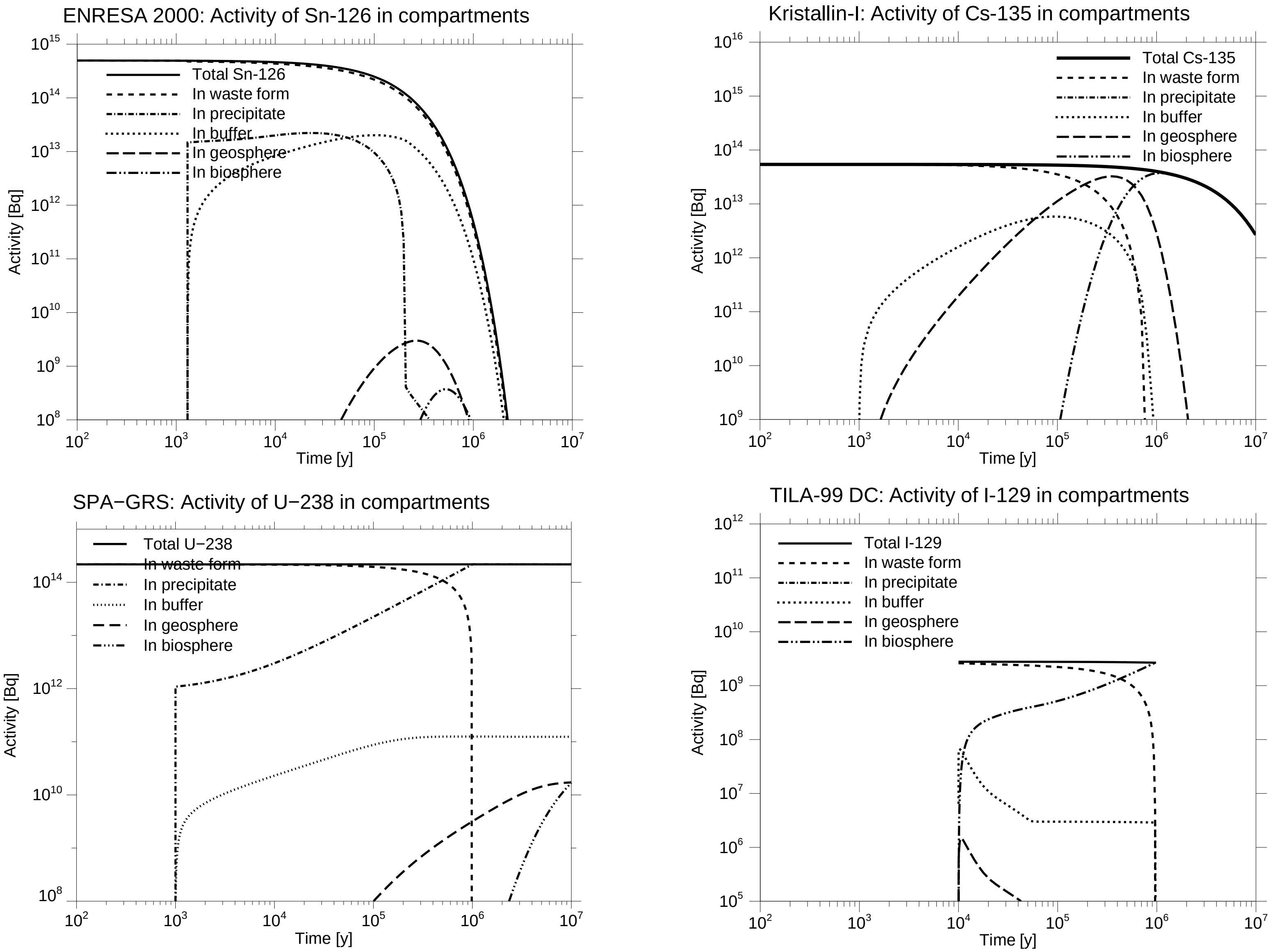

FIGURE 8-3 Activity of specific radionuclides in compartments

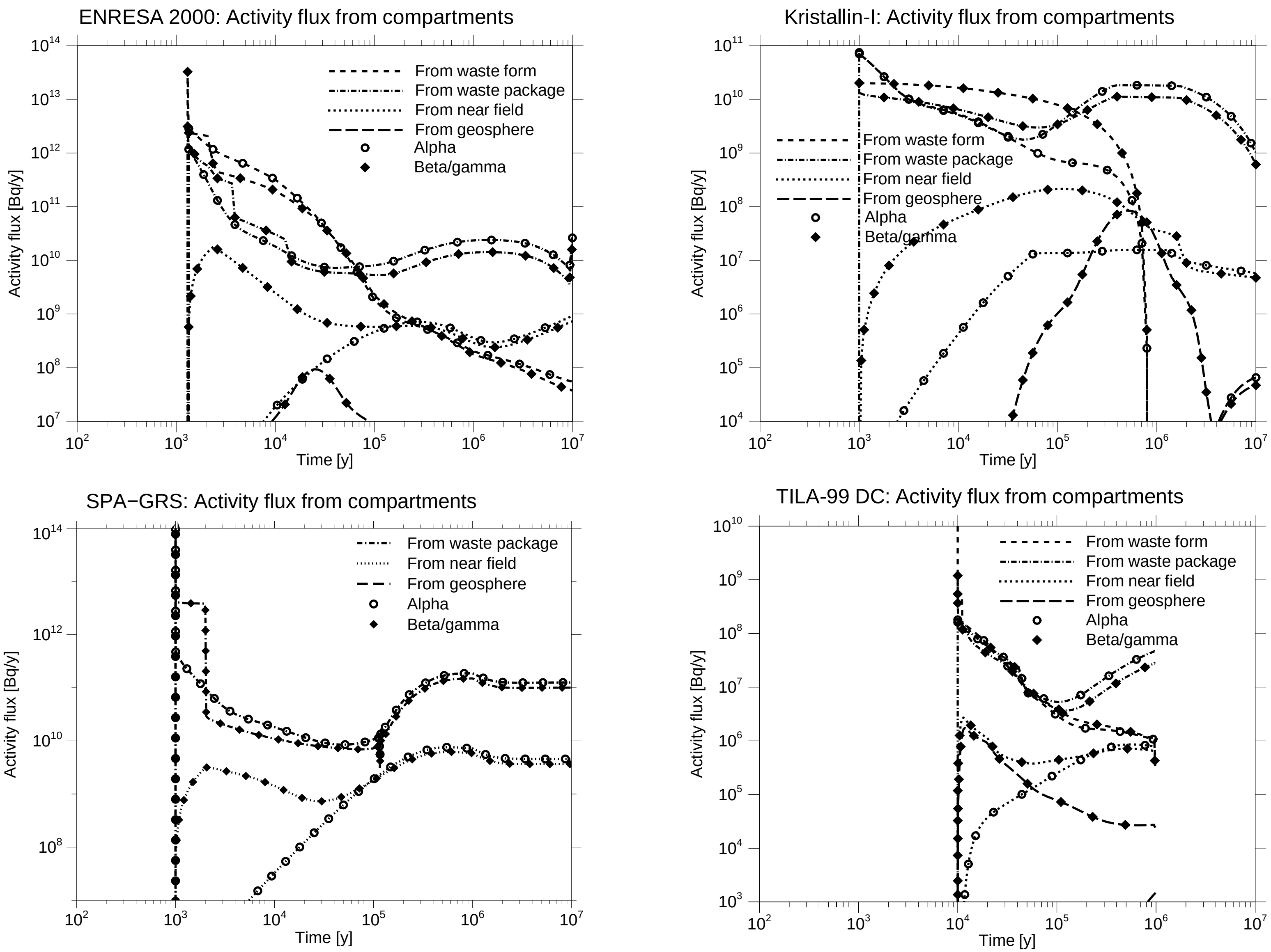

FIGURE 8-4 Total α- and β/γ-activity fluxes from compartments

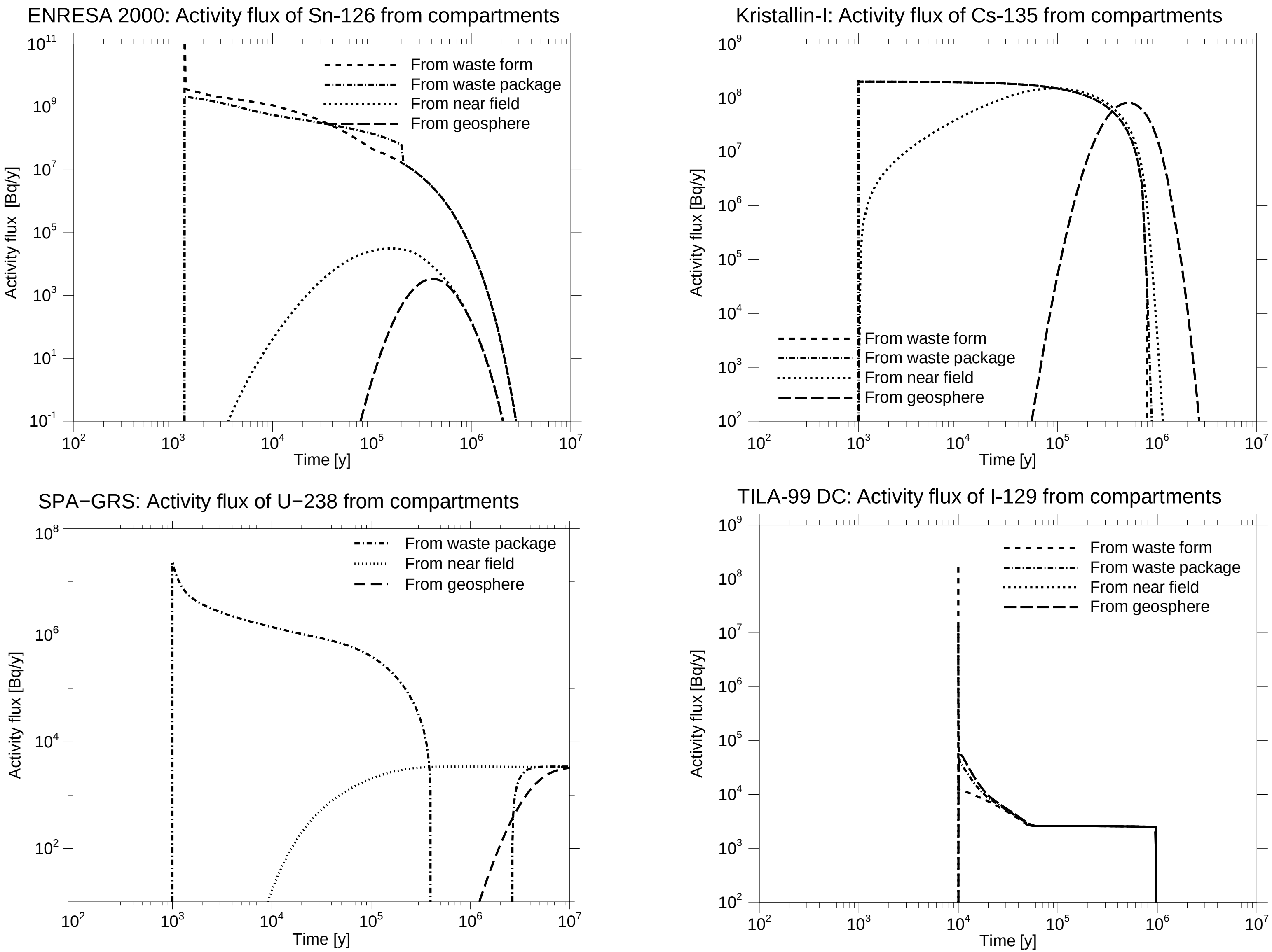

FIGURE 8-5 Activity flux of specific radionuclides from compartments

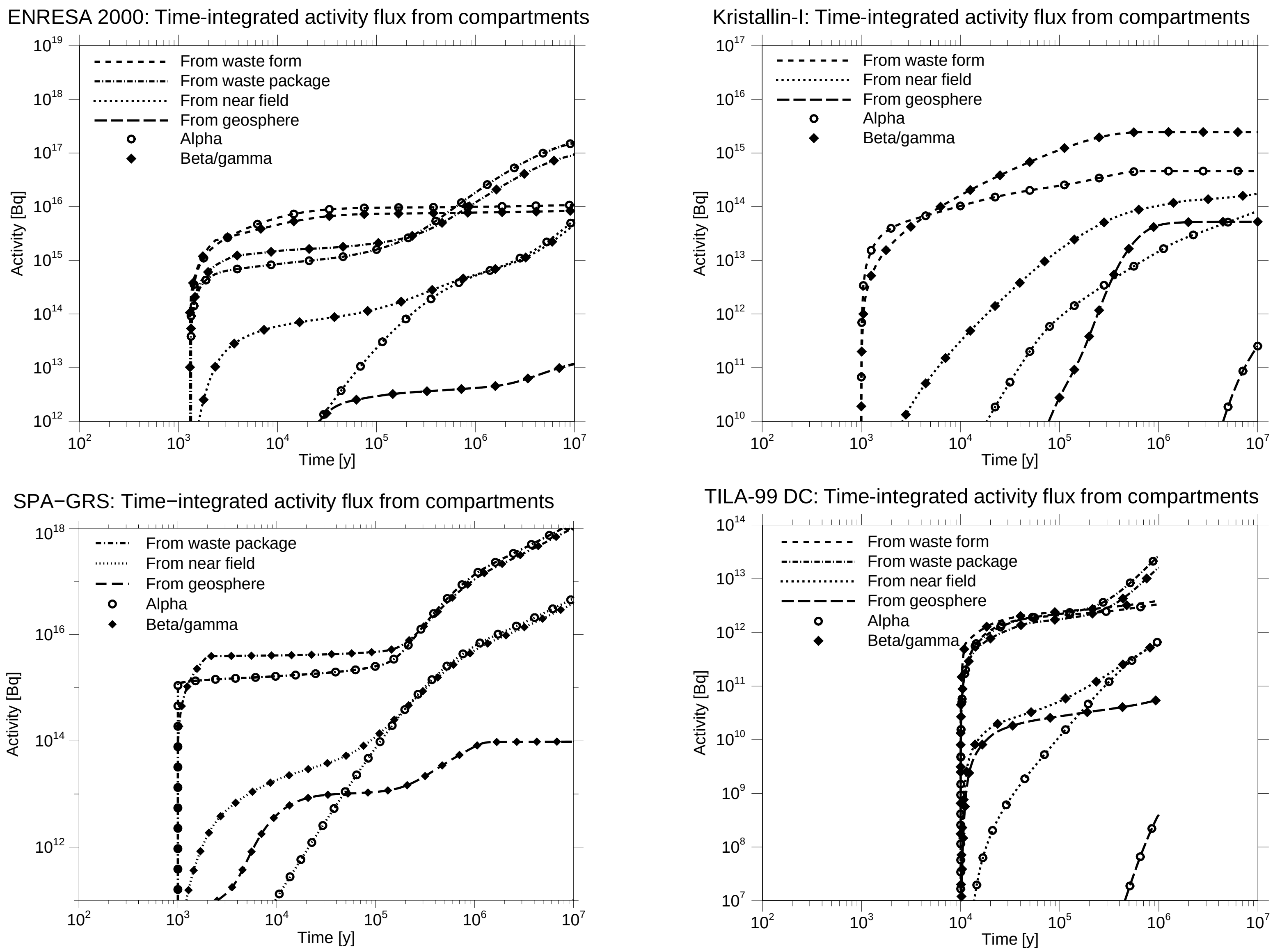

FIGURE 8-6 Total time-integrated α- and β/γ-activity fluxes from compartments

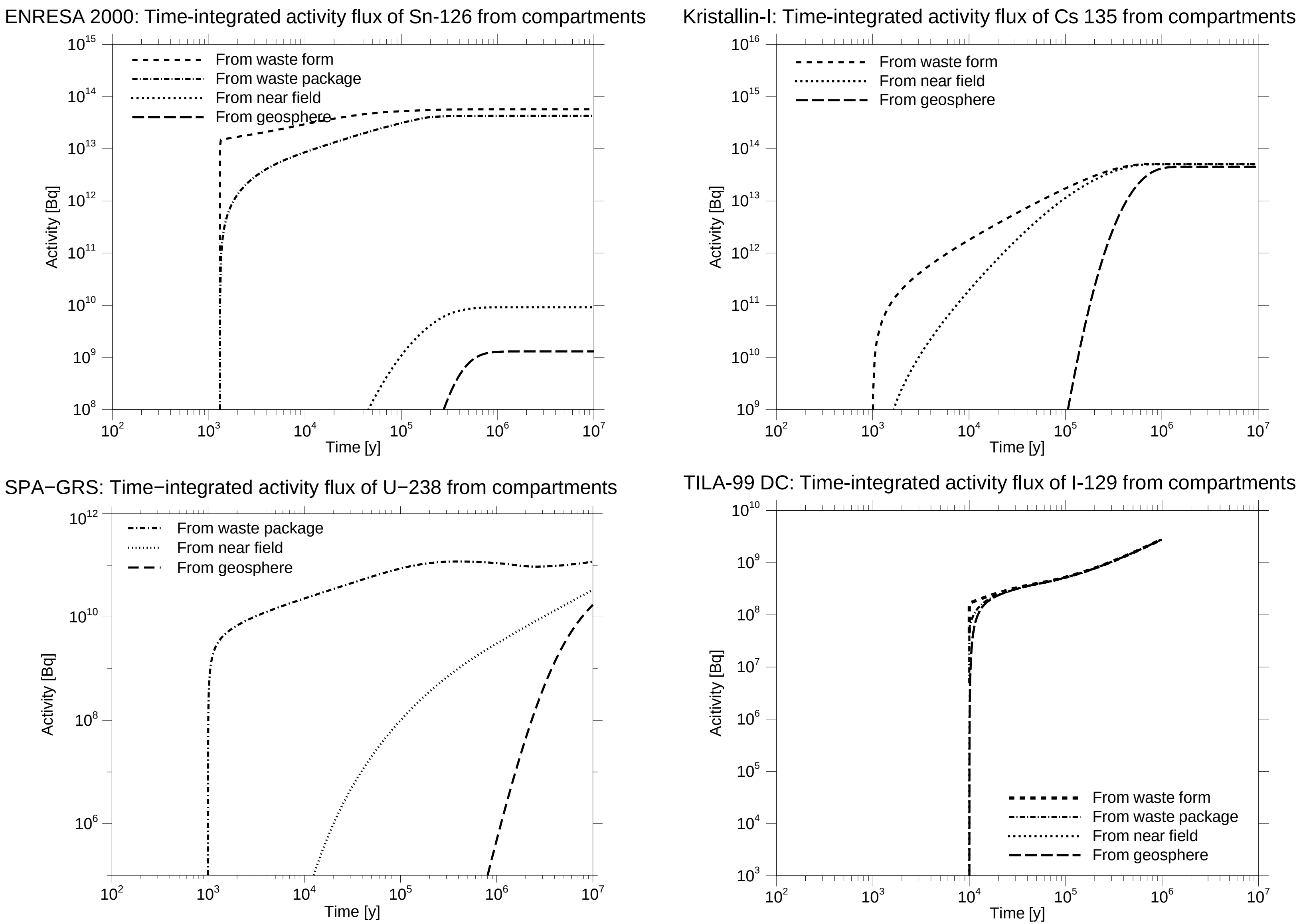

FIGURE 8-7 Time-integrated activity fluxes of specific radionuclides from compartments

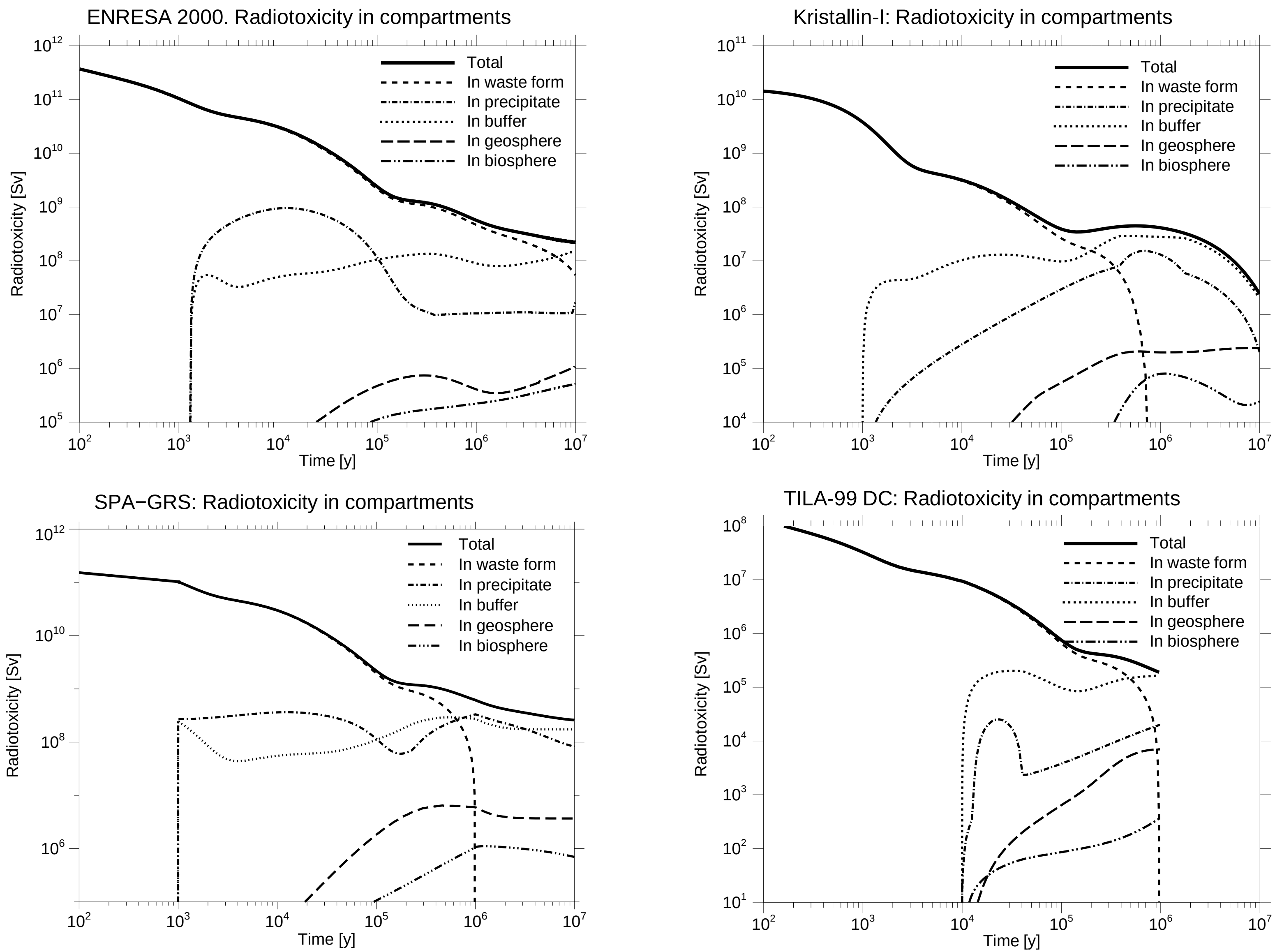

FIGURE 8-8 Radiotoxicity in compartments

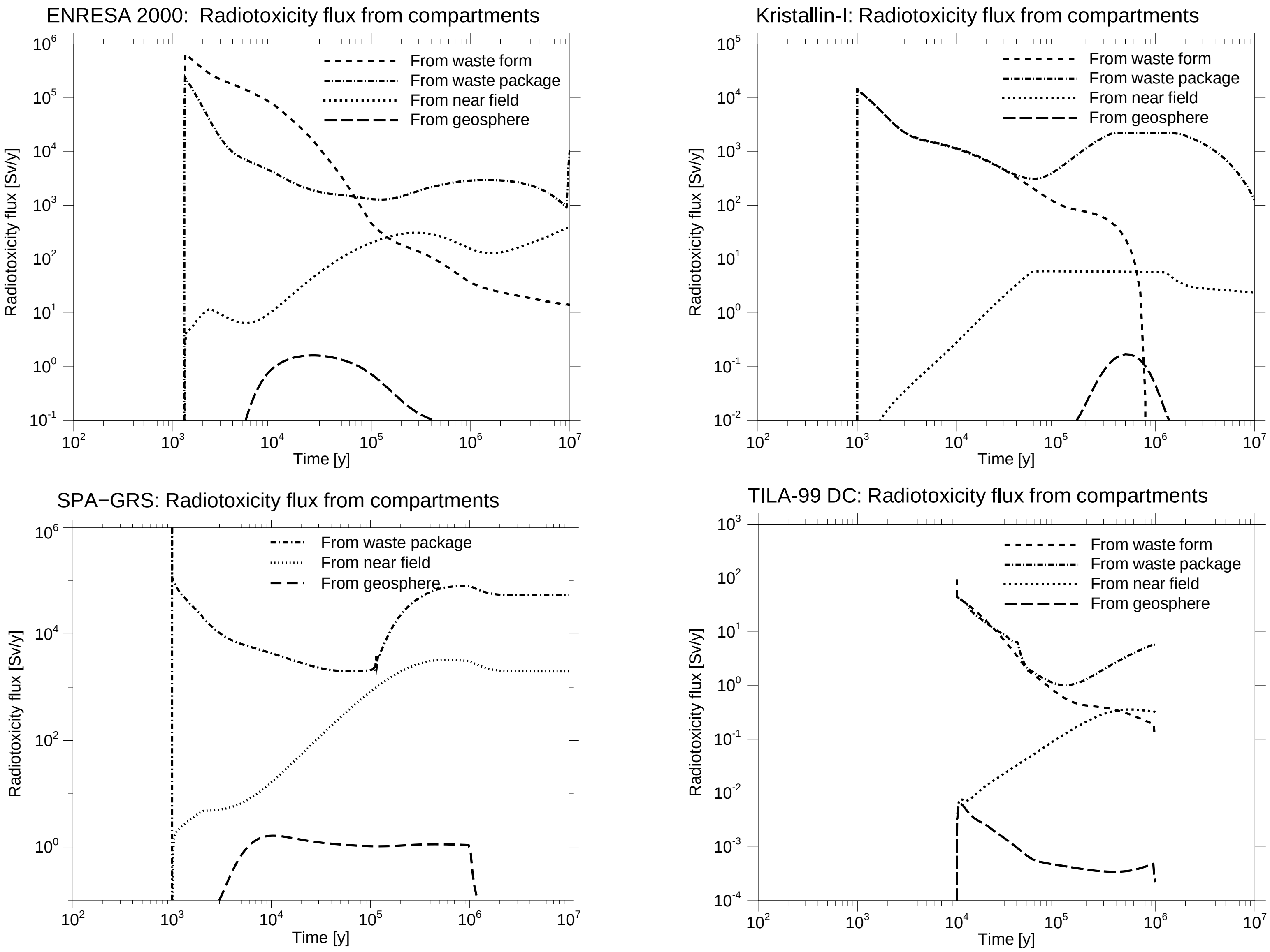

FIGURE 8-9 Radiotoxicity fluxes from compartments

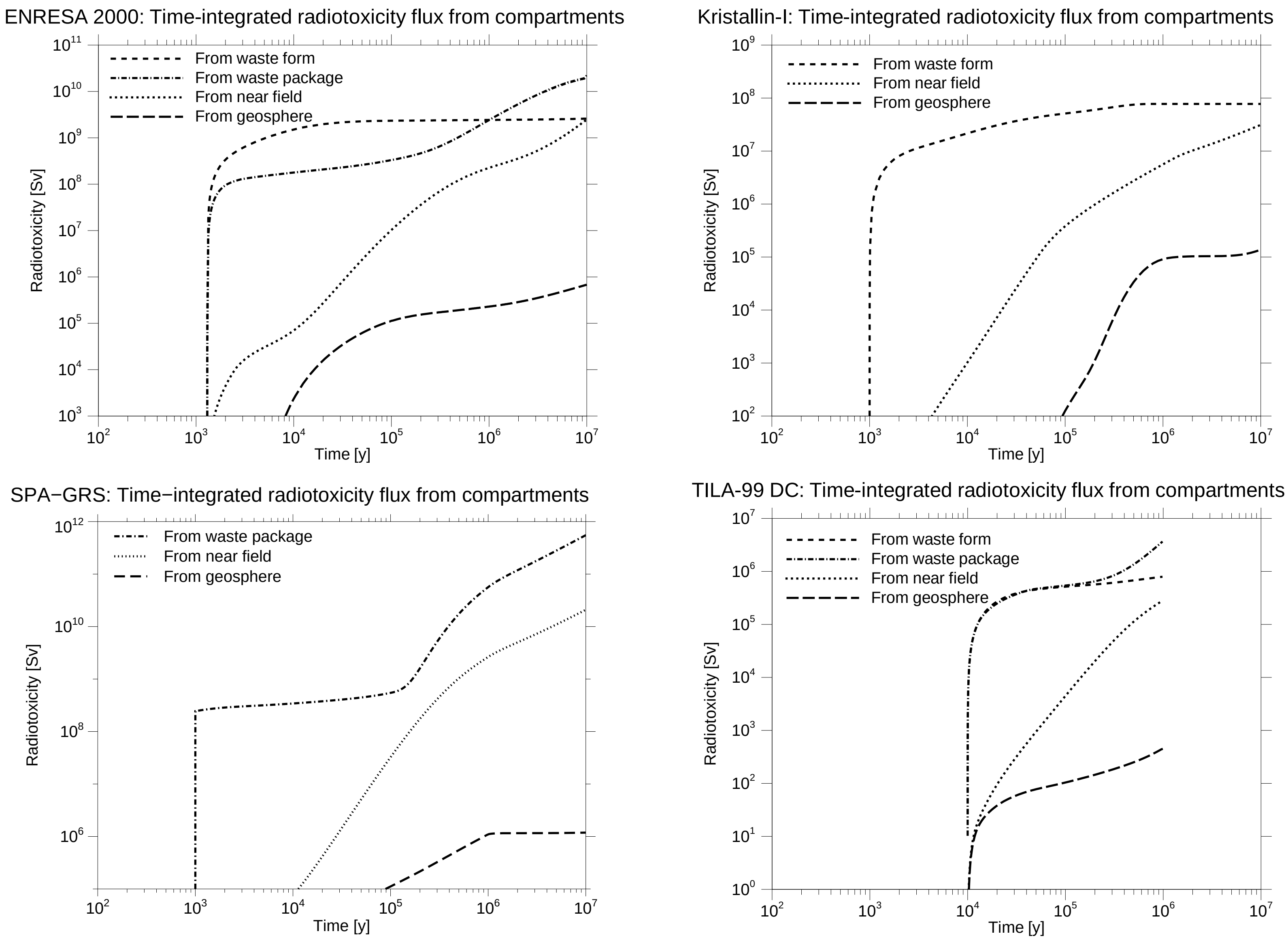

FIGURE 8-10 Time-integrated radiotoxicity fluxes from compartments

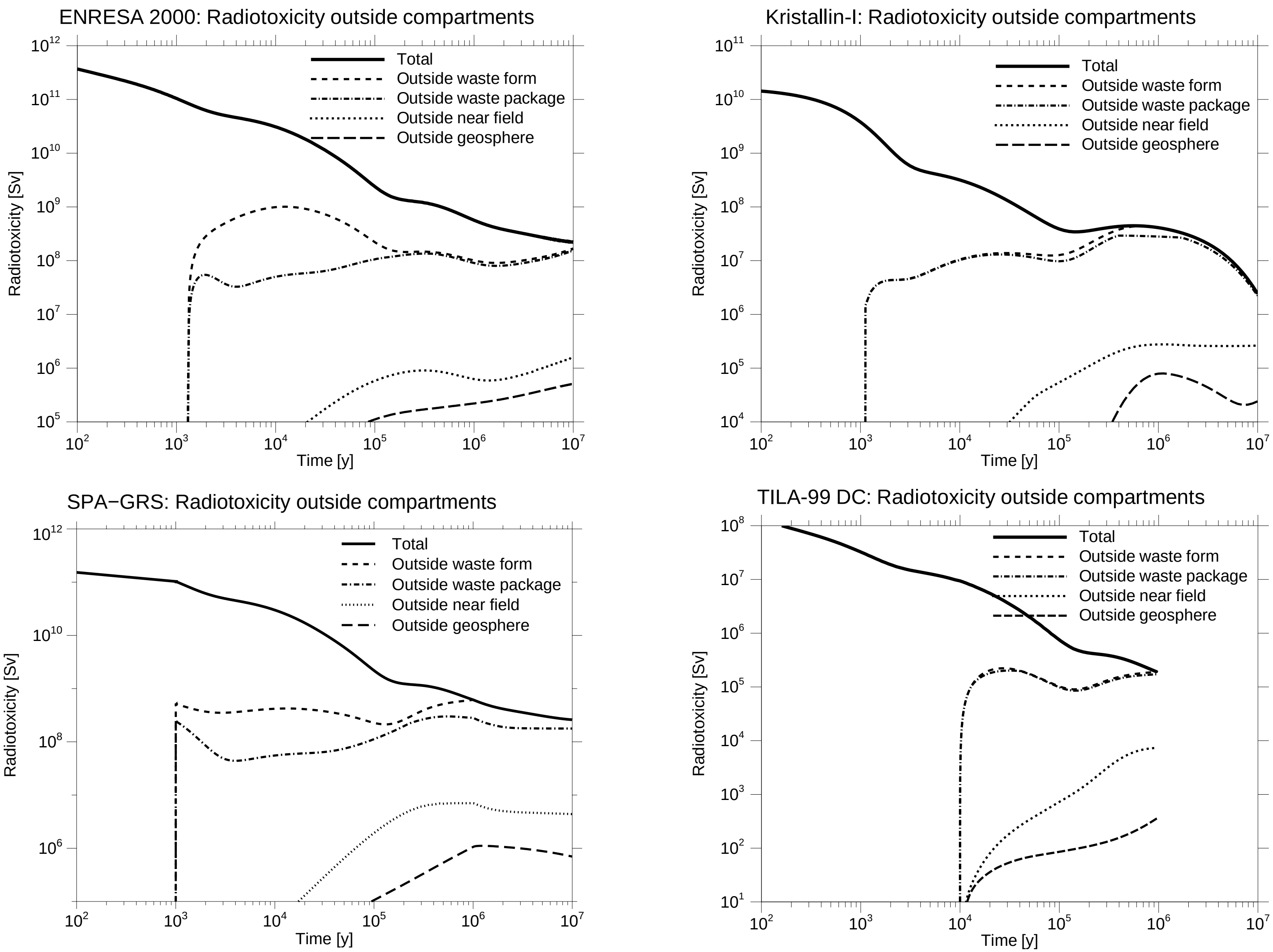

FIGURE 8-11 Radiotoxicity outside compartments

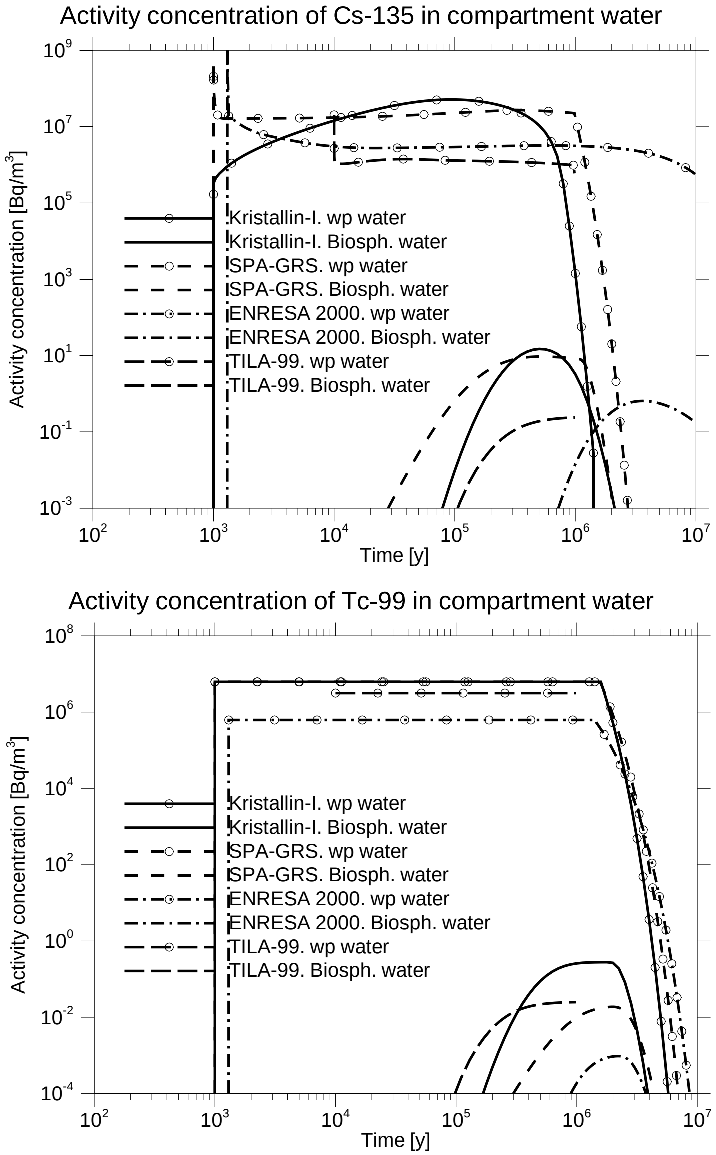

FIGURE 8-12 Activity concentrations in compartment water

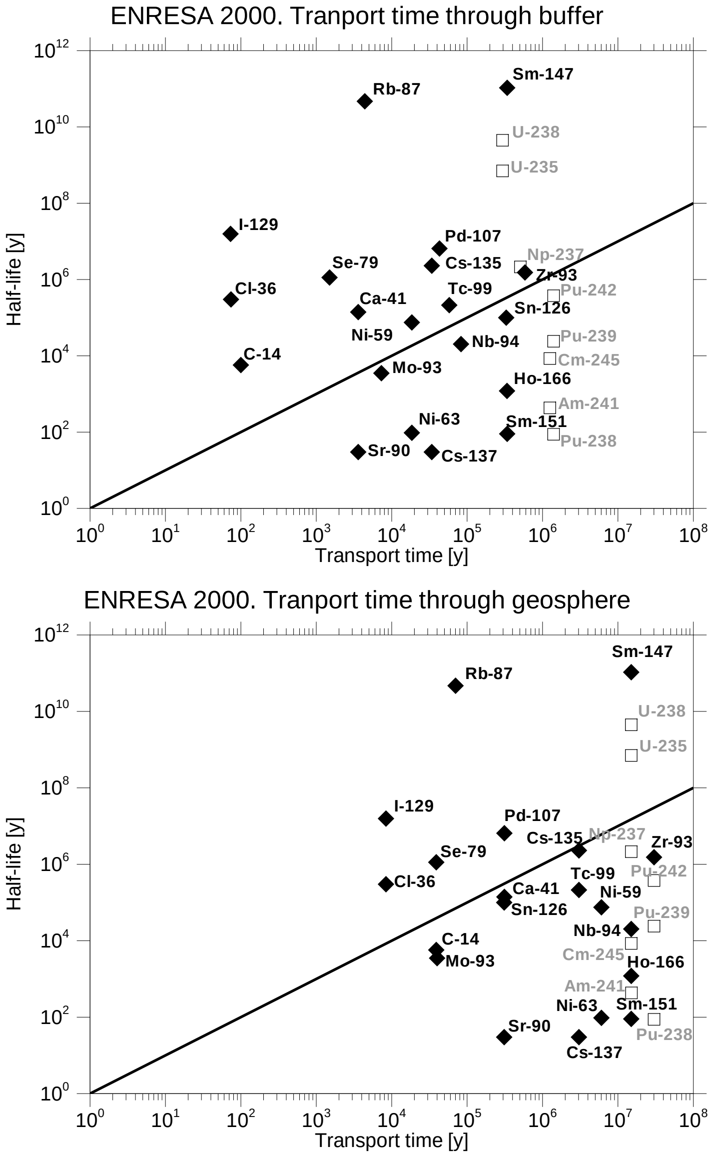

FIGURE 8-13 Transport times through barriers

9 ASSESSMENT OF INDICATORS

In the preceding chapters the calculated results for the selected indicators have been presented. Based on these results each indicator will be assessed on the basis of its specific advantages and disadvantages. This is done in a systematic way by using well-defined basic requirements and additional assessment criteria.

The basic requirements have to be fulfilled by each indicator. They were applied in principle during the selection process and are checked again in the assessment procedure. The assessment criteria judge the usefulness of the indicators and thus facilitate a screening process. Finally, the usefulness of each indicator and the preferred area of application is described.

9.1 Basic requirements for safety indicators

A safety indicator must meet the basic requirements derived from its definition given in Chapter 3. They are compiled in TABLE 9-1. To allow a comparison with safety-relevant reference values such reference values must be available and safety-relevant. To take into account the contributions of all radionuclides a weighting scheme is needed and this must also be safety-relevant.

When radiotoxicity is used for taking all radionuclides into account, the weighting scheme is based on ingestion dose coefficients, which are also included in the biosphere dose conversion factors used to calculate dose rates. These reflect the current state of knowledge and can be considered as safety-relevant. Other suitable weighting schemes can also be used, if they are safety-relevant. A reference value can be automatically included in a weighting scheme, e.g. if nuclide-specific constraints are used.

9.2 Assessment criteria for safety indicators

The criteria used to assess the safety indicators with respect to their practical applicability and meaningfulness in performance assessment calculations are also compiled in TABLE 9-1. Two general criteria consider whether they are easy to understand and whether they provide some added value to others. Two other criteria consider the degree of uncertainty included in the indicators.

Models and data for biosphere pathways include uncertainties which increase significantly with time. Not only the hydrological situation and the climate can change, but also the behaviour of human beings in the future. Indicators which do not rely on assumptions about biosphere pathways therefore have some preference.

Models and data used to establish dilution factors for the aquifer water also include uncertainties which increase with time. Changes of the climate and the geological setting will modify the hydrogeological situation. Also the assumptions about the location for the usage of water include uncertainties. Indicators which do not rely on assumptions about the dilution in aquifer water therefore have some preference.

TABLE 9-1 Requirements and assessment criteria for safety indicators

Categories	Requirements and Criteria
Basic requirements	provide a measure of the safety of the total system
	safety-relevant reference values available
	safety-relevant weighting scheme available
	calculable by use of performance assessment models
Assessment criteria	easy to understand
	added value compared to other indicators
	biosphere pathways excluded
	dilution in aquifer water excluded

9.3 Assessment of safety indicators

The basic requirements and assessment criteria given in the preceding sections were used for the assessment of the safety indicators. TABLE 9-2 gives an overview of the results of the assessment. Plus signs indicate that the criterion is fulfilled, minus signs that it is not. Empty cells indicate that a unique answer cannot be given. The assessment of the indicator *relative activity flux from geosphere* is made on a generic basis that does not take into account the specific features of the flux constraints proposed by the Finnish authorities.

Based on the results of the assessment as presented in TABLE 9-2 another assessment scheme was applied which considered all the advantages and disadvantages of each of the indicators. On this basis, the usefulness of each indicator is assessed and the most appropriate use of the indicator is determined. The assessment of each indicator is presented below.

Effective dose rate

Advantages:
- The basic indicator used to determine the safety of nuclear practices worldwide
- Based on the best safety-relevant weighting scheme for the present biosphere
- Reference values are defined in national regulations

Disadvantages:
- The uncertainties of the biosphere pathways and aquifer dilution are included

Conclusion:
- The indicator is useful for all time frames, but should be given a higher preference for early time frames. The higher preference for early time frames is due to the uncertainties of the biosphere pathways which increase with time

TABLE 9-2 Overview of the results of the assessment of safety indicators

Indicator	Measure for system safety	Reference values		Weighting scheme		Calculable by use of PA models	Easy to understand	Added value	Biosphere pathways excluded	Dilution in aquifer excluded
		available	safety-relevant	available	safety-relevant					
Effective dose rate	+	+	+	+	+	+	+	+	-	-
Radiotoxicity concentration in biosphere water	+	+	+	+	+	+	+	+	+	-
Radiotoxicity flux from geosphere	+	+	+	+	+	+	+	+	+	+
Time-integrated radiotoxicity flux from geosphere	+	+	-	+	+	+	-		+	+
Radiotoxicity outside geosphere	+	+	-	+	+	+	-		+	+
Relative activity concentration in biosphere water	+	-		-		+	+		+	-
Relative activity flux from geosphere	+	-		-		+	+		+	+

Radiotoxicity concentration in biosphere water

Advantages:
- A safety-relevant weighting scheme is available
- The uncertainties of biosphere pathways are excluded
- Safety-relevant reference values can be developed

Disadvantages:
- The uncertainties of the aquifer dilution are included

Conclusion:
- The indicator is useful for all time frames, but a higher preference for early and medium time frames should be given. Because the uncertainties relating to aquifer dilution increases with time this indicator is less relevant for late time frames

Radiotoxicity flux from geosphere

Advantages:
- A safety-relevant weighting scheme is available
- The uncertainties of biosphere pathways and aquifer dilution are excluded
- Safety-relevant reference values can be developed

Disadvantages:
- Reference values can not be established by measurement only: they must be obtained using models, thus introducing another type of uncertainty

Conclusion:
- The indicator is useful for all time frames, but a higher preference for late time frames should be given. The higher preference for late time frames is because the uncertainties relating to biosphere pathways and also aquifer dilution are excluded.

The process of establishing a reference value for this indicator is complicated by the fact that fluxes can not be measured directly but are derived from measured concentrations and assumptions about the relevant hydrogeological setting.

Radiotoxicity outside geosphere and time-integrated radiotoxicity flux from geosphere

Advantages:
- A safety-relevant weighting scheme is available
- The uncertainties of biosphere pathways and aquifer dilution are excluded

Disadvantages:
- Reference values are not safety-relevant
- Not much added value to radiotoxicity fluxes

Conclusion:
- Both indicators are not useful

Reference values for the radiotoxicity outside the geosphere were obtained by integrating the reference values for radiotoxicity fluxes over time. The safety-relevance of this approach is questionable. Reference values derived from the natural radioactivity in biosphere compartments have not been used due to the uncertainty with the definition and the size of safety-relevant biosphere compartments.

Relative activity concentration in biosphere water

Advantages:
- The uncertainties of biosphere pathways are excluded

Disadvantages:
- The uncertainties of the aquifer dilution are included
- A safety-relevant weighting scheme is not available
- Reference values for nuclides not existing in nature are hard to develop

Conclusion:
- The indicator is not applicable unless reference values are found

Relative activity flux from geosphere

The initial assessment of this indicator is made on a generic basis not taking into account the specific features of the flux constraints proposed by Finnish authorities.

Advantages:
- The uncertainties of biosphere pathways are excluded
- The uncertainties of aquifer dilution are excluded

Disadvantages:
- A generally applicable and safety-relevant weighting scheme is not available
- Reference values for nuclides not existing in nature are hard to develop

Conclusion:
- The indicator is not generally applicable unless reference values are found

In addition, the assessment of this indicator is made for the case of Finland, where reference values have been proposed by national authorities. Hence, a country-specific and safety-relevant weighting scheme is available. The constraints are based partly on a set of reference biospheres considered possible in the future at the planned disposal site, and partly on natural fluxes of radionuclides established for similar environments.

The Finnish reference data for activity fluxes are directly applicable only for a certain case in Finland, and cannot be applied universally. Due to the use of biosphere models they incorporate some uncertainties relating to the biosphere pathways. Hence, the added value to dose rates is limited.

9.4 Requirements and criteria for performance indicators

The assessment of performance indicators is less restrictive than that of safety indicators, because performance indicators are used to show the functioning of the system. Hence, the selection of an indicator will depend mainly on the specific problem to be investigated. Nevertheless, some requirements can be defined as given in TABLE 9-3.

TABLE 9-3 Requirements and criteria for performance indicators

Categories	Requirements and Criteria
Basic requirements	measure for the performance of the system or subsystem
	comparison between options or with technical criteria
	weighting scheme available
	calculable by use of performance assessment models
Assessment criteria	easy to understand
	added value compared to other indicators

A key assessment criterion is whether one indicator provides added value compared to others. In general, it was found that each of the indicators provides different, though sometimes similar, information and therefore provides some added value. Therefore, depending on the specific situation and depending on the intended investigation, a selection from the range of indicators should be made.

9.5 Assessment of performance indicators

The requirements and criteria given in the preceding section were used for the assessment of the performance indicators. A compilation of major advantages and disadvantages is given below for each indicator, together with some conclusions concerning their recommended use.

When investigating the performance of sub-systems or the total system taking account of all radionuclides, it is often not meaningful to sum activities of different radionuclides, as their radiotoxicity level varies. A weighting scheme defined by the ingestion dose coefficients can be used to define integral performance indicators based on radiotoxicity. Performance indicators based on activities are better used for single nuclides. Such indicators allow to compare how a given system works for different types of nuclides or, conversely, how different sub-systems work for the same nuclide.

Activity and radiotoxicity in compartments

Advantages:
- Clear and easy to understand
- Show where the nuclides are
- Allow an assessment of the retention capabilities of barriers

Disadvantages:
- The value in biosphere can become relatively high, which may be misleading.

Conclusions:
- Useful to show where the radionuclides are at each instant
- Useful to show the functioning of the multi barrier system

Activity and radiotoxicity flux from compartments

Advantages:
- Clear and easy to understand
- Show the rate at which the nuclides move from compartment to compartment

Disadvantages:
- None

Conclusions:
- Useful to show decreasing release rates from compartment to compartment
- Useful to show the barrier function of the disposal system

Time-integrated activity and radiotoxicity flux from compartments

These indicators can be calculated in absolute terms or normalised to the initial inventory at repository closure.

Advantages:
- Clear and easy to understand when applied to single radionuclides
- Show the retention capabilities of each compartment
- Independent of the biosphere model and dilution

- The values after 'infinite' time show how much has decayed inside each compartment

Disadvantages:
- Only increasing with time, up to large absolute values which can be misleading
- Not easy to understand when applied to the total radiotoxicity from all radionuclides

Conclusions:
- Useful to show the decay during delayed transport
- The normalised versions should be used, they provide more meaningful information

The *time-integrated flux from a compartment* represents the amount of radionuclides that escaped from that compartment up to a given time, regardless of the possible radioactive decay of these radionuclides after they left the compartment. Hence, the interpretation as a quantity outside a compartment is not meaningful. If the integrated flux is compared to the initial inventory in the waste, it provides an indication of the level of isolation provided by a part of the barrier system.

For individual radionuclides, these indicators allow to quantify the fraction of the initial inventory that decays in each compartment if considered for sufficiently long time periods. However, if the indicator is applied on the basis of total radiotoxicity, fluxes from inner compartments can be dominated by short-lived decay products of high toxicity, and thus the value of the indicator can exceed the initial inventory. Meaningful results, however, can be achieved for time-integration of the radiotoxicity flux from geosphere, where the effect of such short-lived daughters is generally small.

Activity and radiotoxicity outside compartments

Advantages:
- Clear and easy to understand
- Show the combined performance inside a compartment boundary
- Show which barriers are ineffective

Disadvantages:
- The added value to 'inventory in compartments' is difficult to understand

Conclusions:
- They can be used as an alternative to 'inventory in compartments'
- Useful to show the overall effectiveness of components of the disposal system

Activity and radiotoxicity concentration in compartment water

Advantages:
- If calculated for waste package and biosphere, it shows the total reduction of radionuclide concentration caused by the disposal system

Disadvantages:
- The concentrations in buffer and geosphere water are space-dependent and therefore not well-defined

Conclusions:
- Useful to show contributions of dispersion, dilution and decay phenomena acting together
- Useful to show the barrier function of the disposal system

The concentrations in some compartments such as the buffer and the geosphere are dependent on space. To use well-defined concentration only, this indicator is limited to the waste package water and the biosphere water. In some cases also the water of the excavation damaged zone in the rock around the buffer material can be considered.

The reduction in concentrations as described by this indicator are not only caused by dispersion and dilution phenomena but also by the time history of the release processes and the radioactive decay.

Transport time through compartments

Advantages:
- Illustrative and easy to understand
- Gives an overview of the potential nuclide importance

Disadvantages:
- Requires a well-defined calculation scheme
- Provides only a rough measure of the importance of radioactive decay
- Limited to considerations on individual nuclides

Conclusions:
- Useful to identify unimportant and potentially important radionuclides, with respect to safety

The transport time through compartments and also through the total barrier system can be compared to the half-life of individual radionuclides. This comparison provides a qualitative orientation of which individual radionuclides are of no importance and which are of potential importance for long-term safety. For a quantitative information on the importance of the decay during the transport and for an integral view comprising all radionuclides other indicators, the *time-integrated activity and radiotoxicity fluxes from compartments*, can be used.

Proportion of not totally isolated waste

Advantages:
- Characterises the safety function 'isolation and physical containment'

Disadvantages:
- Difficult to be defined for different types of waste

Conclusions:
- Useful to show the safety function 'isolation and physical containment'

In many disposal systems a proportion of the waste may be considered to be completely isolated for a given time period, e.g. because it is in long-lived containers or in dry parts of the host rock. During the relevant time period there is no release of radionuclides from the waste form.

Concentration in biosphere water divided by concentration in waste package water

Advantages:
- Characterises the dispersion and dilution process in the barrier system.
- Excludes radioactive decay

Disadvantages:
- Conservative estimate only, because a constant concentration in the waste package water is considered

Conclusions:
- Useful to show the safety function 'dispersion and dilution'

This indicator is applied for a stable nuclide and for a constant concentration in the waste package water. By this procedure the time history of the release from the waste form and the radioactive decay will not be taken into account and the result shows only the effect of dispersion and dilution.

10 CONCLUSIONS

To contribute to the development of additional indicators other than dose and risk, several indicators which can be calculated from performance assessment results were tested. Seven safety indicators and fourteen performance indicators were considered, using the following basic definition: a safety indicator is concerned with the level of long-term safety of the disposal system as a whole and a performance indicator is concerned with the functioning of subsystems or of the total system with respect to radionuclide retention.

The indicators have been tested by re-calculating existing performance assessments of disposal systems for high level waste in crystalline formations. Although other geological formations have not been tested, the conclusions of the project might be more generally applicable.

In the project the effective dose rate is taken as the basic safety indicator, since its use is already well established and it enables a direct assessment of the impact of a disposal system on human health. Two other indicators were found to provide significant benefits and may therefore be used to complement the effective dose rate. In summary, the three proposed safety indicators and their preferred application time frames are:

- *Effective dose rate*: most relevant to early time frames
- *Radiotoxicity concentration in biosphere water*: preference for medium time frames
- *Radiotoxicity flux from geosphere*: preference for late time frames.

Each of the three indicators can and should be applied to all time frames, but there are preferences. For early time frames, as which the first several thousand years are considered, the effective dose rate is preferred. It includes the representation of all the biosphere pathways and is therefore the most relevant indicator during this time frame. The biosphere pathways include an increasing level of uncertainty over longer time periods. Therefore the effective dose rate should be given less preference for medium and late time frames.

The radiotoxicity concentration and the radiotoxicity flux do not take biosphere pathways into account and are therefore better applicable to medium and late time frames. For medium time frames, as which the time period from several thousand to several tens of thousands of years is considered, the radiotoxicity concentration is preferred. It includes the dilution in the aquifer and the river systems as they are today and is therefore the most relevant indicator during medium time frames. For later times even the dilution gets uncertain and the radiotoxicity flux becomes the most relevant indicator for late time frames of hundreds of thousands of years and more.

Safety indicators require reference values as a basis for comparison. For the effective dose rate the data from present regulations were taken and used as a range of reference values. Reference values for radiotoxicity concentration and fluxes can be determined by considering levels of radionuclides in the natural environment, based on the assumption that nature in general is safe from a radiological point of view. Data for radionuclide concentrations and fluxes in a range of crystalline environments were used as a basis for developing reference values..

The radiotoxicity flux itself excludes the uncertainties of the dilution and is therefore largely independent of the hydrogeological situation in the aquifer. The development of the corresponding reference value, however, often takes the hydrogeological situation into account. Since fluxes can not be measured directly they are often derived from measured concentrations and assumptions about the present-day hydrogeological situation. In this case the radiotoxicity flux as an indicator compared to its reference value does not provide any added value compared to the radiotoxicity concentrations.

For demonstrating the performance of the repository's multi-barrier system, the project concluded that several indicators can be used to show different aspects of the functioning of the individual barriers and the multi-barrier system. These indicators and their preferred applications are:

- *Inventories in compartments:* showing where the radionuclides are at different points in time,
- *Inventories outside compartments:* showing the retention capability of all inner barriers,
- Fluxes from compartments: showing the decreasing release rates from successive compartments, including radioactive decay and ingrowth,
- *Concentrations in compartment water:* showing the increasing dilution in successive compartments,
- *Transport times through compartments:* showing the potential importance of individual radionuclides to the release of radiotoxicity by comparing them to their half-lives.

For investigations relating to the total radionuclide spectrum, performance indicators based on radiotoxicities should be used. Performance indicators based on activities are considered appropriate for individual radionuclides, when investigating the different behaviours of the different types of radionuclides.

The indicators given above can be identified by considering the basic safety functions of the repository's multi-barrier system and can also be used to demonstrate how these safety functions are fulfilled. Three other performance indicators have been developed to show exclusively the safety functions 'physical isolation', 'decay during the delayed transport' and 'dispersion and dilution':

- Proportion of waste not completely isolated for a given time period: physical isolation function,
- *Time-integrated flux from compartments divided by initial inventory:* decay during delayed transport function,
- *Concentration in biosphere water divided by concentration in waste package water:* dispersion and dilution function.

These indicators have been defined in such a way that low numbers indicate a good performance of the disposal system.

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A1 IDENTIFICATION OF PERFORMANCE INDICATORS ON THE BASIS OF FUNCTIONAL REQUIREMENTS OF THE REPOSITORY SYSTEM

It is internationally accepted that the basic strategy for the management of high-level radioactive waste should be to 'concentrate and contain' radionuclides rather than to 'dilute and disperse' them in the environment [35]. Consequently, the primary objective of geological disposal is to isolate the radionuclides as long as possible from the human environment. However, because isolation cannot be guaranteed over all time, geological disposal must also delay and reduce as far as possible any eventual release of radionuclides into the environment.

To realise the above objectives geological repository systems consisting of various engineered and natural barriers are being developed. A safety assessment aims at systematically analysing the performance of the integrated repository system. Such an assessment comprises the identification of scenarios that represent the various possible evolutions of the repository system as well as the many interactions between the various components, and evaluations of the system performance for the selected scenarios.

The functioning of a multi-barrier repository system can be explained in simple terms by defining a set of functional requirements [15], [36]. These requirements can be used in communication with technical and non-technical audiences to explain how the repository system works, for assessing the contribution of a single barrier to the performance of the integrated system, and for deriving design criteria for engineered barriers. One or more barriers of the repository system contribute to the realisation of each of these functional requirements.

A1.1 Functional requirements

Four basic functional requirements, which correspond to the expected normal functioning of the repository system, can be identified: physical containment, slow release, retardation, and dispersion and dilution. The primary objective of geological disposal, i.e. isolation from man and the environment, is realised by the three first functional requirements. During this isolation period, the activity of many radionuclides present in the disposed waste will considerably decrease by radioactive decay. The second objective 'delay and reduce' is mainly realised by the three last functional requirements.

The functioning of a repository system can be disturbed by the evolution of its components or by the occurrence of low-probability events caused by natural phenomena or future human actions. Two complementary functional requirements contribute to limiting the rate of those evolutions and the impact of low-probability events: stability of the engineered and natural barriers, and limitation of access; these complementary requirements are not considered in the SPIN project.

A1.1.1 Physical containment

This functional requirement refers to the isolation of the radioactive waste in a watertight barrier during the first phase after repository closure. As long as this requirement is effective, there is no contact between groundwater and the radionuclides present in the disposed waste, and, consequently, no release of radionuclides can occur from the waste form.

Physical containment makes the disposal system more robust and easier to analyse by preventing dispersion of radionuclides during the strongly transient initial phase of the repository history (resaturation processes, heat release, strong β/γ radiation, pressure rebuilding, etc.). At the same time it allows to take advantage of the radioactive decay of the short-lived radionuclides.

In most repository systems for high-level waste disposal, this functional requirement is performed by a long-lived metallic container. The longevity of the container can be based on the wall thickness (carbon steel), the very slow corrosion rate (stainless steel), or the thermodynamic stability of the material (copper). The bentonite buffer indirectly contributes to this functional requirement by creating a favourable chemical environment around the container.

A1.1.2 Slow release

After container failure, when groundwater comes in contact with the conditioned waste, leaching of radionuclides from the waste matrix starts in combination with the degradation of the waste matrix. Various physico-chemical processes, such as lixiviation from and corrosion of the waste matrix, precipitation, sorption on or co-precipitation by secondary phases, strongly limit the radionuclide releases into the surrounding layers. The waste containers presumably will not fail all at the same time; this will result in a time-spread at least for the instant release fraction (e.g. gap / grain boundary inventory in the case of spent fuel elements) of the radionuclide inventory. In case of a partial failure of the container (e.g. pinhole), radionuclide releases will be limited by geometrical constraints.

A consequence of this functional requirement is that even after the perforation of the containers most radionuclides will stay during very long times within the volume of the waste package; further benefit is made of the radioactive decay of the radionuclides. The slow release also drastically limits the amounts of radionuclides that are released from the waste package per unit of time.

The main barriers that contribute to this functional requirement are the waste matrix and the precipitate, the latter is a result of the low solubility of many radionuclides. As long as the defects of a container are limited to one or a few holes, the perforated container will also contribute to this requirement.

A1.1.3 Retardation

The radionuclides dissolved in the groundwater that is in contact with the waste will start to migrate through the bentonite buffer. Because of its very low hydraulic conductivity the transport through the buffer will be mainly diffusive. Furthermore, many radionuclides will be sorbed onto the clay minerals of the buffer material. During the advective transport through the host rock, matrix diffusion can further retard the radionuclide transport.

Retardation drastically limits the amounts of long-lived radionuclides that can appear in the biosphere per unit of time. The very long travel times through the bentonite buffer and the host formation allow to take maximum benefit of the radioactive decay of the longer-lived radionuclides.

For deep disposal in crystalline rock, the main barriers that contribute to this functional requirement are the bentonite buffer and the host rock formation.

A1.1.4 Dispersion and dilution

Once the long-lived radionuclides leave the host rock, they are, considerably scattered in time, released into the surrounding aquifers and eventually into the accessible environment. The dispersion and dilution processes in the aquifers and surface waters will further reduce the radionuclide concentrations in the waters that are directly accessible by man.

Dispersion and dilution can only be of secondary importance because any attempt to maximise or optimise this functional requirement would lead to a 'dilute and disperse' strategy, instead of the intended 'concentrate and contain' strategy.

The components that fulfil this functional requirement are the less robust of the repository system; indeed, the hydrogeological and the hydrological surroundings of the repository can be considerably affected by various disturbing factors (climatic changes, geomorphologic processes, human impacts, etc.).

A1.2 Performance indicators

Within the consequence analyses of the performance assessment of the repository system various additional output variables can be calculated which illustrate the functioning of the repository system and which quantify the contribution of each of the functional requirements. Such output variables are called performance indicators [19]. Performance indicators can be used to demonstrate the contributions of the above mentioned basic functional requirements. An overview of the identified indicators is given in TABLE A1-1.

A performance indicator for physical containment is the estimated lifetime of the watertight container. The robustness of this functional requirement depends on the amount and quality of scientific evidence that the required lifetime can readily or adequately be ensured. Within the SPIN project the container lifetime was considered as a design parameter and no specific performance indicators related to this functional requirement have been developed.

Direct indicators of the performance of the 'slow release' and 'retardation' functional requirements in combination with the 'physical containment' are the time-dependent activity content of each compartment [22], and the flux of activity that is released from each compartment. The cumulative fraction of the activity of a radionuclide that is released from each compartment and that eventually reaches the biosphere illustrates how successful is the contribution of the first three basic functional requirements in combination with radioactive decay.

The transport or breakthrough times of radionuclides through important physical barriers (e.g. bentonite buffer, geosphere) are practical indicators for quantifying the capability of the disposal system to delay and, hence, to limit radionuclide releases. These indicators in comparison with the half-lives of safety-relevant radionuclides provide valuable information on the efficiency of the corresponding transport barrier as a consequence of radioactive decay.

Dilution can be quantified by an apparent dilution rate which is defined as the ratio of the maximum flux released from the host rock over the maximum concentration calculated in the aquifer from which man might pump groundwater. The apparent dilution rate is mainly determined by the amount of groundwater flowing through the aquifers above, or under, the repository area together with smaller contributions of dispersion and sorption processes. In the

SPIN project the dilution occurring in the repository system is quantified by the indicator *concentration in biosphere water divided by concentration in waste package water*. However, it should be noted that the releases of radionuclides from the host formation are expected to occur several thousands to hundreds of thousands of years after disposal and that the hydrogeological environment in the far future can be very different from the present one, mainly as a result of climate changes (greenhouse effect, glaciation, etc.).

TABLE A1-1 Basic functional requirements of a repository system and related performance indicators

<table>
<tr><th>Functional requirement</th><th>System component ('safety barrier')</th><th>Performance indicator 1)</th></tr>
<tr><td>Physical containment</td><td>(Intact) container</td><td>Container lifetime [y]</td></tr>
<tr><td rowspan="3">Slow release</td><td>Failed container
- distribution of failure time
- pin hole (restricted release)</td><td>'Time reduction factor'
'Geometrical reduction factor'</td></tr>
<tr><td>Glass matrix / SF matrix (incl. cladding)
- dissolution / corrosion</td><td rowspan="4">Dissolution rate [1/y] or time [y]

Activity in compartments [Bq]
Activity flux from compartments [Bq/y]
Time-integrated activity flux from compartments [Bq]
Radiotoxicity in compartments [Sv]
Radiotoxicity flux from compartments [Sv/y]
Time-integrated radiotoxicity flux from compartments [Sv]
Radiotoxicity outside compartments [Sv]

Transport time through compartments [y]
- bentonite buffer
- geosphere transport barrier</td></tr>
<tr><td>Precipitate
- solubility limit</td></tr>
<tr><td rowspan="2">Retardation</td><td>Bentonite buffer
- diffusion
- sorption</td></tr>
<tr><td>Geosphere
- transport (advection / diffusion / dispersion)
- sorption
- matrix diffusion</td></tr>
<tr><td rowspan="2">Dispersion/ Dilution</td><td>Geosphere
- flow rate
- path length
- sorption / matrix diffusion</td><td rowspan="2">Activity concentration in compartment water [Bq/m^3]:
- near field and biosphere
Radiotoxicity concentration in compartment water [Sv/m^3]:
- near field and biosphere
Concentration in biosphere water / concentration in waste package water

In addition for geosphere:
'apparent dilution factor'= $\frac{\text{max. RN flux from host rock}}{\text{max. RN concentration in aquifer}}$</td></tr>
<tr><td>Biosphere
- aquifer (or river)
- well
- top soil (sorption, accumulation)</td></tr>
</table>

1) Indicators that have been tested within the SPIN project are indicated in italics

A1.3 Conclusions

A clear and transparent explanation of the functioning of a disposal system is a prerequisite for communicating the results of complicated safety evaluations of deep disposal systems to larger audiences. For that purpose, a systematic analysis of the disposal system in terms of the basic functional requirements can be very helpful.

Starting from four basic functional requirements, namely physical containment, slow release, retardation, and dispersion and dilution, various performance indicators can be defined, which allow to quantify the contribution of each of the different barriers to the total performance of the repository system.

A2 IDENTIFICATION OF PERFORMANCE INDICATORS ILLUSTRATING THREE BASIC SAFETY FUNCTIONS

A repository for the disposal of radioactive waste in geological formations should isolate the disposed waste as safely as possible from the biosphere. To reach this goal a multi-barrier design for the repository system is used. A series of individual barriers contribute in different ways to the overall performance and hence to the safety of the repository system. In the case of the disposal of high-level waste in a granite formation such barriers are the waste form, the container, the bentonite buffer and the geosphere.

The individual barriers contribute by different features to the safety and the performance of the repository system. Performance indicators for the multi barrier system can be derived through a systematic approach using safety functions. A safety function is a specific feature of the barriers or a specific process in the barrier system which contributes to the safety. The following basic safety functions are considered here:
- isolation or physical containment for all the time under consideration,
- delay of radionuclide release to the biosphere and radioactive decay during that time,
- dispersion and dilution of radionuclides in the near field and far field.

Isolation or physical containment in this context means the total isolation of a fraction of the disposed waste in the host formation. The radionuclides in this fraction will never start to move towards the biosphere. If radionuclides from the remaining part of the waste start moving towards the biosphere, the slow and delayed transport in connection with the radioactive decay will reduce the amount of radionuclides ever reaching the biosphere. For those radionuclides moving towards the biosphere the dispersion and dilution will distribute them in space and time and will reduce their concentrations.

The approach to performance indicators described in the following is aimed at identifying just one performance indicator for each of the three safety functions, i.e. avoiding multiple performance indicators for one safety function. Performance indicators should not lump several functions or processes which could be kept separate, so that any overlap between the performance indicators will be avoided. This will result in performance indicators which are as independent as possible.

The three safety functions will be explained in detail and three performance indicators representing the three safety functions will be identified. The performance indicators will be defined in a similar manner to allow for an easy interpretation. The definition of release fractions is used here, so that low numbers always indicate a good performance.

A2.1 Physical Containment

The physical containment is the total isolation of all the waste or a fraction of the waste in the host formation over all the time under consideration. An example are containers with very long life times for example over millions of years. Another example is waste in low permeable formations which could be completely surrounded by dry host rock. The radionuclides in these wastes will never start moving towards the biosphere as long as the stability of the host formation is maintained.

The performance indicator for this safety function is the proportion of waste which is not totally isolated in the host formation and therefore has to be considered in the assessment. The value of this indicator can be a given parameter derived from assumptions about the number of failed containers or the amount of waste to be included in the assessment over the time period to be considered in the assessment. The value of this indicator can also be developed within the modelling, when the number of waste packages getting into contact with water is derived within the calculation scheme.

A2.2 Delay and Decay

That part of the waste which is getting into contact with water as a transport medium and which is defined by the first indicator starts releasing radionuclides towards the biosphere. It will take some time for the radionuclides to reach the biosphere and during this time the radioactive decay will reduce the amount of radionuclides.

The transport will be retarded by each of the barriers. The container will remain watertight for some time and delay the start of the release from the waste form. The resistance of the waste form against its degradation as well as the precipitation of parts of the radionuclides will spread the release from the waste form over a long period of time. The time for diffusion through the bentonite buffer and the time for advective transport through the geosphere will give more time for further decay. The sorption of radionuclides on the buffer material as well as the diffusion into and the sorption on the rock matrix will increase the transport times.

The performance indicator for this safety function is the proportion of radionuclides that reach the biosphere. The amount of radionuclides reaching the biosphere has to be related to that part of the disposed waste, which has the potential of being transported towards the biosphere. The performance indicator can be calculated by accumulating the release from the geosphere over the time period to be considered compared to the initial amount of radionuclides which are not totally isolated in the host formation.

The indicator can be established on the basis of activities for individual radionuclides and on the total radiotoxicity of all radionuclides. Besides application to the release from the geosphere it can also be applied to the release from more inner barriers. The consideration of released radiotoxicities from more inner barriers might cause difficulties at the interpretation of results, because of the contribution of the accumulated fluxes of daughter products. Hence, the consideration of releases from the geosphere is preferred.

A2.3 Dispersion and Dilution

The dispersion and diffusion occurs along the transport pathways from the disposed waste to the biosphere. The diffusion of radionuclides in the pore water of the transport pathway and dispersion in the porous medium distribute the radionuclides over a larger space and into a larger water body. In particular the release of contaminated water into an aquifer or river with fresh water increases the volume of the water body for the radionuclides. All these processes result in a reduction of radionuclide concentrations. Reduced radionuclide concentrations will decrease the radiation doses to future human beings.

The performance indicator for this safety function is the reduction factor for the concentration in the waste package water compared to the concentration in the biosphere water provided by the diffusion, dispersion and dilution processes only. For the dissolved concentration in the waste package water a constant value is used. The concentration in the biosphere water to be used is the maximum value after a sufficiently long time. To exclude the decrease of the concentration due to decay the calculations have to be made without radioactive decay.

Due to the fixed concentration in the waste package water and the neglect of radioactive decay the calculated results represent the dispersion and dilution phenomena only. The limited release time from the waste package will give an additional reduction in the concentration of the biosphere water. This effect is not included in the performance indicator since the release time for solubility controlled radionuclides might be very long. The effect of radioactive decay is not included in this performance indicator since it is completely included in the indicator for delay and decay.

A2.4 Summary

By considering the three safety functions of a repository system also three performance indicators have been identified. These are given in TABLE A2-1. Each of the performance indicators fully represents the respective safety functions and none of them does include any overlap with the others. The performance indicators are defined in a similar manner to allow for an easy interpretation: low values of a performance indicator always shows a good performance.

TABLE A2-1 Safety Functions and Performance Indicators

Safety Function	Performance Indicator
Physical confinement	waste not totally isolated in the host formation divided by total amount of waste in the repository
Delay and decay	time-integrated flux from the geosphere divided by initial inventory of not totally isolated waste
Dispersion and dilution	concentration in biosphere water divided by concentration in waste package water

For the indicator of physical containment the total amount of waste in the repository and that part which is not totally isolated in the host formation can be given in tons or m^3 if just one type of waste is to be considered. If several types of waste are disposed in the repository a more sophisticated quantity such as the radiotoxicity has to be applied.

For the indicator of delay and decay the integration of the flux from the geosphere can be presented as a function of time. In the long term an asymptotic value will be reached in general. In this case the indicator can be given as a single value which allows for an easy interpretation and an easy comparison with the first indicator. This indicator can be quantified using activities of individual radionuclides and also radiotoxicities.

For the indicator of dispersion and dilution the concentration in the waste package water is presented in an arbitrary unit. The concentration in the biosphere or aquifer water will be calculated as a function of time. After a sufficiently long calculation time an asymptotic value will be reached in general. In this case the indicator for can be given as a single number which allows for an easy interpretation and an easy comparison with the other indicators.

As a supplement to this report, a CD-ROM is available that contains, apart from an electronic version of the report, a large number of coloured figures presenting the calculation results in detail. To receive a copy of the CD-ROM, please write to

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