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Project SAFE

Update of the SFR-1 safety assessment Phase 1

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October 1998

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Abstract

This report contains the results of the first phase of project SAFE (Safety Assessment of Final Repository for Operational Radioactive Waste). The aim of project SAFE is to update the previous safety analysis of SFR-1 and to prepare a safety report that will be presented to the Swedish authorities not later than in year 2000. SFR-1 is a facility for disposal of low-level radioactive waste, which is situated in bedrock beneath the Baltic Sea, 1 km off the coast near the Forsmark nuclear power plant in northern Uppland.

The first phase of project SAFE is a prestudy with the aim to identify issues where additional studies would improve the basis for an updated safety analysis as well as to suggest how these studies should be carried out. The observations and recommendations compiled in this report are the result of a re-evaluation of the previous safety assessment and the supporting reports, taking the comments and remarks given by the authorities in their review of the previous safety reports into account.

Contents

1 Introduction

1.1 Background and Aim

SFR-1 is a facility for disposal of low-level radioactive operational waste from the nuclear power plants in Sweden. Low-level radioactive waste from industry, medicine, and research is also disposed in SFR-1. The facility is situated in bedrock beneath the Baltic Sea, 1 km off the coast near the Forsmark nuclear power plant in northern Uppland. SFR-1 was built between the years 1983 and 1988.

An assessment of the long-term performance of the facility was included in the vast documentation that was a part of the application for an operational license. The assessment was presented in the form of a final safety report (SSR, 1987).

The report was presented to the Swedish authorities (SSI, the Swedish Radiation Protection Institute, and SKI, the Swedish Nuclear Power Inspectorate) in 1987. License for operation was granted in March 1988, but with the requirement of a complementary analysis of some specific issues. The result of the complementary analysis was presented in a deepened safety report in 1991 (FSA, 1991).

In the operational licence for SFR-1 it is stated that renewed safety assessments should be carried out at least each ten years. In order to meet this demand SKB has launched a special project, SAFE (Safety Assessment of Final Disposal of Operational Radioactive Waste). The aim of the project is to update the safety analysis and to prepare a safety report that will be presented to the Swedish authorities not later than in year 2000.

Project SAFE is divided into three phases. The first phase is a prestudy, and the results of the prestudy are given in this report. The aim of the prestudy is to identify issues where additional studies would improve the basis for the updated safety analysis as well as to suggest how these studies should be carried out.

1.2 Outline of the Report

Safety assessment is a complex activity that covers many scientific fields and questions. Therefore the work has been divided into six different topics, namely the inventory, the near field, the far field, the biosphere, radionuclide transport calculations and scenarios. For each topic the former safety reports and regulatory reviews are scrutinised and needs for additional work is identified. The evaluations are given in appendices covering the respective topics. Since many questions are of interest in more than one aspect there are overlaps between the different appendices.

The main report is a summary of the appendices with a more stringent description of the repository system and the processes that are of interest and therefore should be addressed in an updated safety assessment. However, it should be pointed out that one of the improvements proposed as a part of the prestudy concerns the development of a systematic description of the SFR process system (see Chapter 6 and Appendix A2). Such an exercise may reveal additional improvements or new issues of importance for the safety assessment.

An overview of the repository site and repository design is given in Chapter 2. In Chapter 3 the previous safety assessments are summarised together with comments made in the regulatory reviews of the previous assessments. In Chapter 4 the result of reviewing the earlier description of the operation of the repository and possible incidents during the operational phase is given. Identified needs of improvements concerning the information on waste inventory and waste allocation are summarised in Chapter 5. In Chapter 6 a proposed approach to scenario and system analysis is outlined. Identified needs of new analysis of topics related to the evolution of properties and conditions in the near-field, far-field and biosphere are summarised in Chapter 7 and improvements concerning the prediction of radionuclide transport and radiological impact are summarised in Chapter 8. Finally, some recommendations for the next phases of project SAFE are given in Chapter 9.

The appendices to this report are published in a separate report, SKB Report R-98-44, Swedish Nuclear Fuel and Waste Management Co.

2 Repository Design and Site Description

2.1 General

SFR-1 is situated in Forsmark in the northern part of Uppland, close to the Forsmark nuclear power plant. The storage vaults are located in the bedrock, approximately 60 m below the seabed, 1 km off the coast. The underground part of the repository is accessed through two tunnels.

SFR-1 is designed for final disposal of low and intermediate level radioactive waste from the Swedish nuclear power plants and the Central Interim Storage for Spent Nuclear Fuel (CLAB) and for similar waste from other industries, research and medical care.

In total the SFR-1 is intended for 90 000 m^3 of waste. In the previous safety assessment the total radioactivity in this waste was assumed to be 10^{16} Bq (SSR, 1988). Today (1998) the waste capacity in the existing parts of the facility is approximately 60 000 m^3 and approximately 23 000 m^3 of waste is disposed.

There are plans for a further expansion of the facility in a later stage, in order to dispose of radioactive waste from the decommissioning of the nuclear power plants (SFR-3). This expansion will require a new permit by the Swedish government since it is not covered in the existing permits.

The ongoing pre-study of the Östhammar area has identified an area about 1 km from the SFR-1 site, which could be suitable for a deep repository. No decision about further investigations or construction of a repository has been made.

The repository is designed to isolate the waste from the biosphere in order to avoid harmful consequences to man and environment both under operation and after closure. This is accomplished by the emplacement in rock under the seabed and by the technical barriers surrounding the waste.

SFR-1 is divided into four types of rock vaults:

- The Silo
- Rock Vault for Intermediate Level Waste (BMA)
- Rock Vault for Concrete Tanks (BTF)
- Rock Vault for Low Level Waste (BLA)

The rock vaults are connected through a system of tunnels. The design of the first stage of SFR-1 is shown in Figure 2-1. There are also plans for a second stage of SFR-1 containing mainly the addition of some BMA-vaults. The consequences of these potential vaults will be addressed within project SAFE.

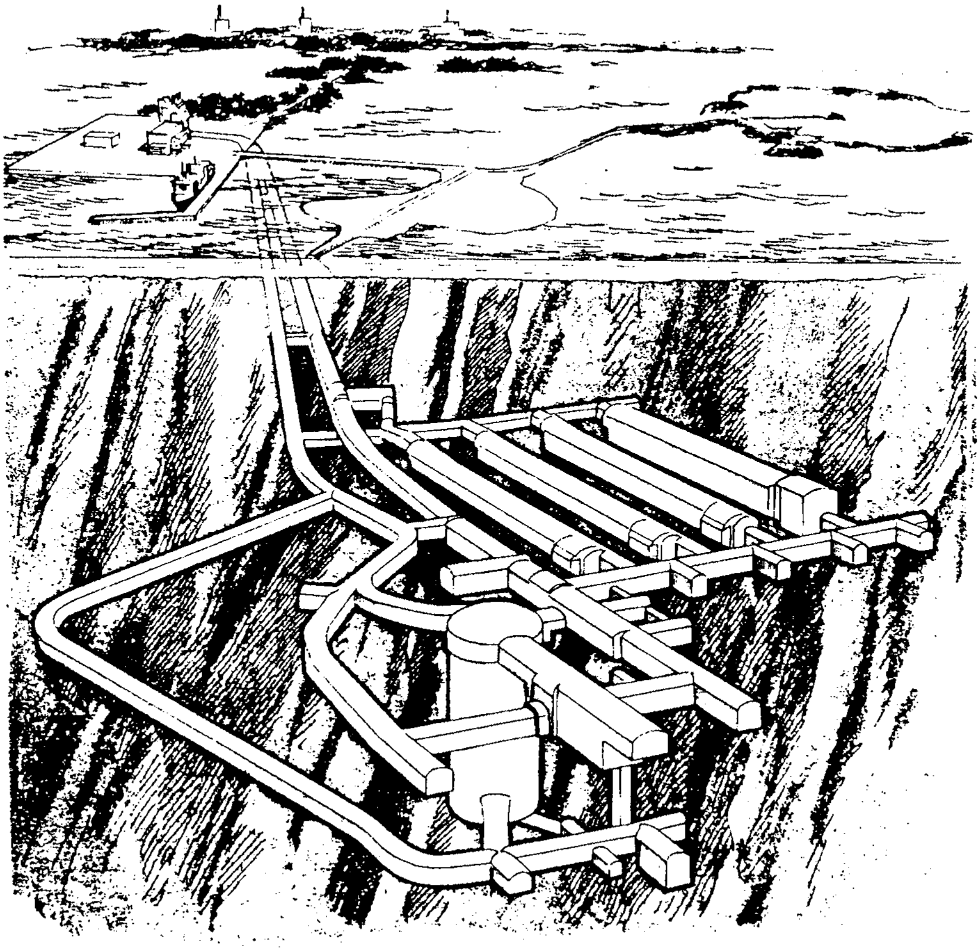

Figure 2-1. The first stage of SFR-1

2.1.1 The Silo

The main part of the radioactivity in the waste for SFR-1 is intended for disposal in the Silo. The Silo is designed for approximately 18 500 m^3 of conditioned waste. The waste is mainly composed of ion-exchange resin in a concrete or bitumen matrix. It is also foreseen that a small part of scrap metal and trash will be deposited in the Silo, although at present the authorities approve no waste of that kind. The waste is normally packed in concrete or steel moulds of the dimensions 1.2x1.2x1.2 m or in 200 litre steel drums.

The Silo consists of a concrete cylinder with the height of approximately 50 m and a diameter of approximately 30 m, see Figure 2-2. The Silo is divided into vertical shafts with intervening concrete walls. The waste packages are placed in the shafts, normally in levels with four moulds or 16 drums. The voids between the waste packages are gradually backfilled with porous concrete.

The walls of the Silo are made of reinforced concrete with a thickness of 0.8 m. In between the Silo walls and the surrounding rock there is a bentonite backfill, on average

1.2 m thick. The 1 m thick concrete floor in the bottom of the Silo is placed on a layer of 90/10 sand/bentonite mixture. According to present plans the top of the Silo will be a 1 m thick concrete lid with gas vents through the lid to allow gas to escape from the Silo. The top of the lid will be covered with a thin layer of sand and then 1.5 m of sand/bentonite mixture (90/10). The remaining void above the sand/bentonite in the top will be filled with sand or gravel or sand stabilised in cement.

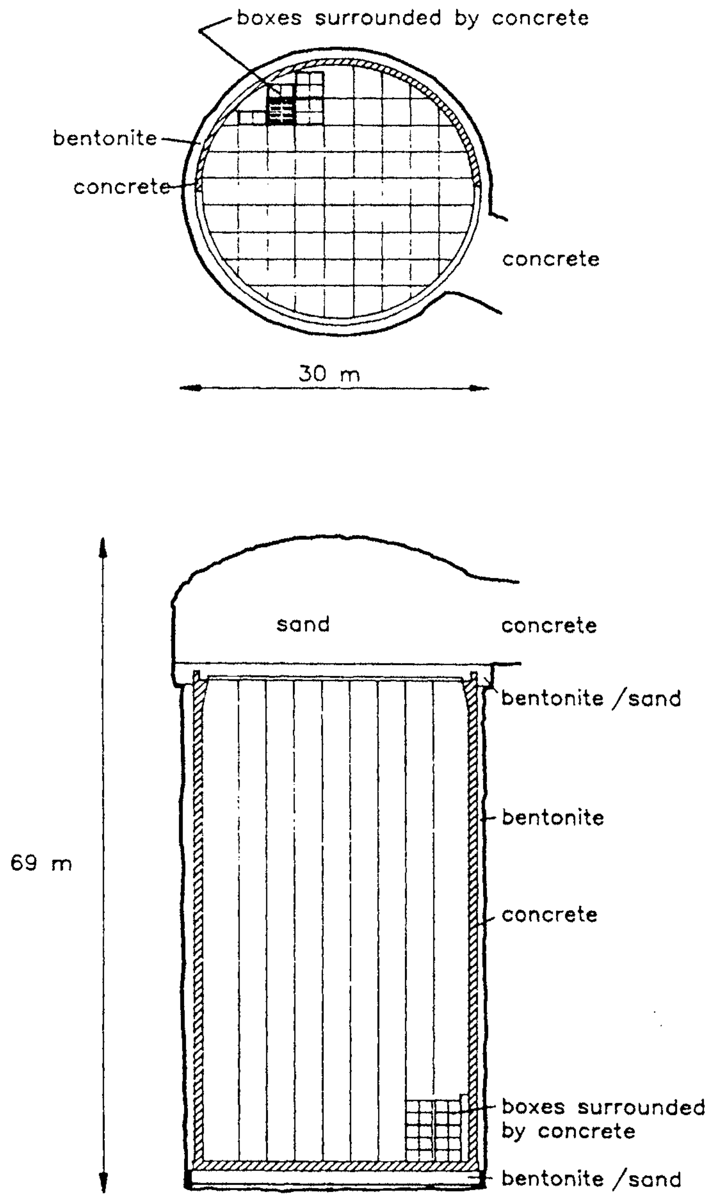

Figure 2-2. The Silo

2.1.2 The rock vault for intermediate level waste (BMA)

The radioactivity in the waste that is deposited in the BMA is mainly lower than in the waste in the Silo. The waste consists of ion exchange resins, scrap metal and trash in a concrete or bitumen matrix. The waste packages are of the same type as in the Silo, i.e. moulds and drums.

The rock vault is 160 m long and has a cross-section of approximately 300 m^2, see Figure 2-3. The concrete structure in the vault is divided into 15 compartments with a total effective volume of 13 400 m^3 for conditioned waste. The compartments are built like big boxes with concrete walls in between. The waste is piled on top of the concrete floor in a way that allows the concrete moulds to act as support for pre-fabricated concrete lids. The lids are put in position as soon as the compartments are filled. Finally a layer of concrete is casted on top of the lid.

Between the concrete structure and the rock wall there is a 2 m wide space. This space will be filled with sand. The space above the concrete structure will be left mostly unfilled. Plugs will probably be placed in the two entrances to the vault when the repository is closed.

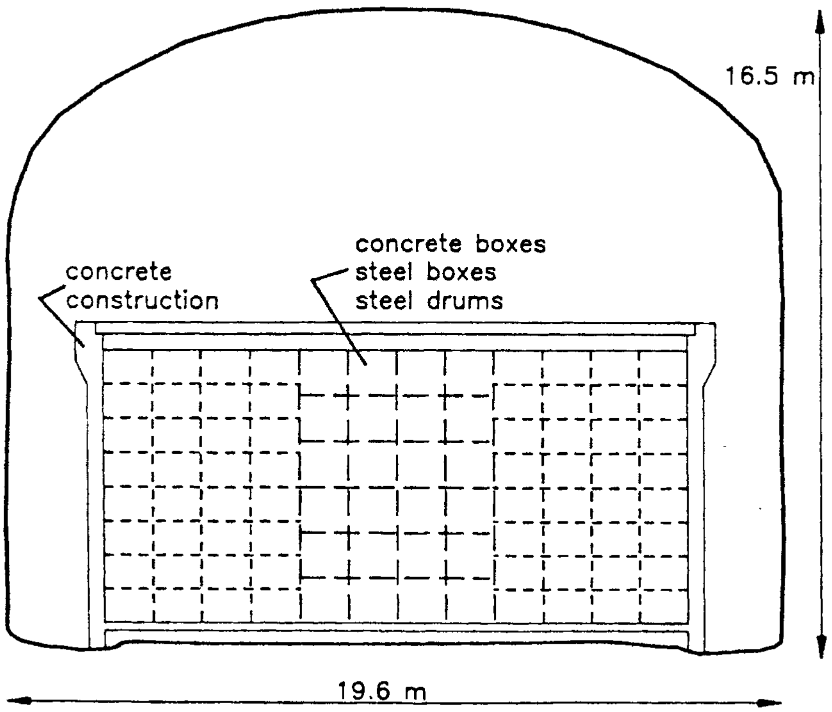

Figure 2-3. The BMA

2.1.3 The rock vault for concrete tanks (BTF)

In SFR-1 there are two rock vaults for concrete tanks, 1BTF and 2BTF. The effective waste volume is 7900 m^3. The waste placed in BTF is de-watered low-level ion exchange resin in concrete tanks. In addition, some drums with ashes are stored in 1BTF.

The rock vaults are 160 m long and has a cross-section of approximately 130 m^2, see Figure 2-4. The vaults contain no concrete structures except for the floor.

The concrete tanks, each with a volume of 10 m^3, are piled in two levels with four tanks in each row. A concrete radiation protection lid is placed on top of the pile. The space between the different tanks is backfilled with concrete and the space between the tanks and the rock wall will be filled with, for example, sand stabilised in cement. The space above the lids is mostly left unfilled.

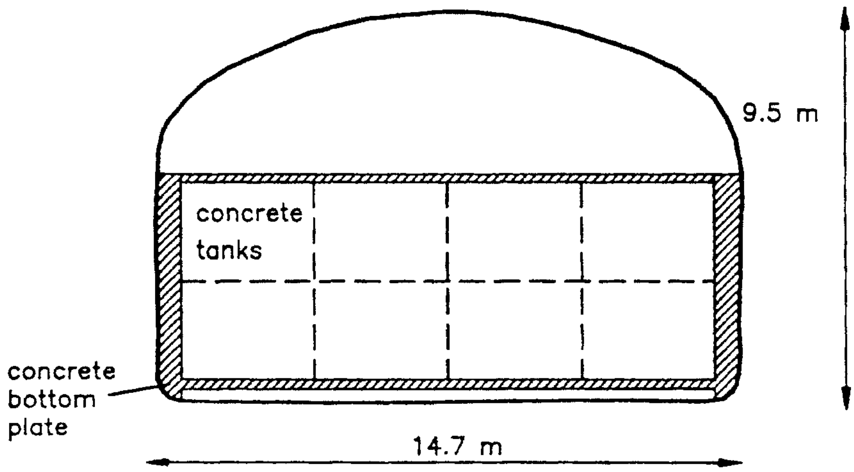

Figure 2-4. 1BTF and 2BTF

2.1.4 The rock vault for low level waste (BLA)

The waste deposited in BLA is mainly low level trash placed in standard steel containers. The effective waste volume is approximately 11 500 m^3. Some of the waste inside the containers is placed in steel drums and others in bales.

The rock vault is 160 m long and has a cross-section of approximately 180 m^2, see Figure 2-5. There are no concrete structures in BLA except the floor.

The containers are placed two in a row and three full height containers in height. Most of the containers are half height and these are piled to a height of six. No backfill is planned.

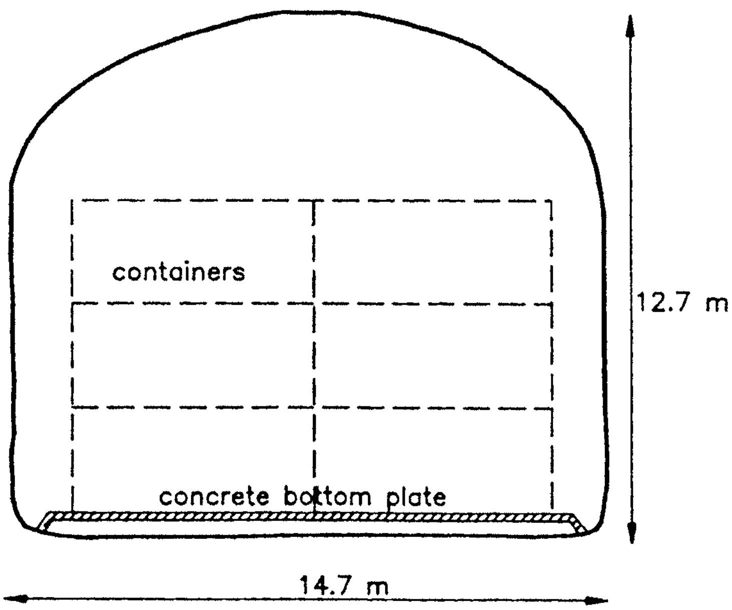

Figure 2-5. The BLA

3 Previous Work and Regulatory Reviews

3.1 Previous Work

This section contains a short summary of the topics addressed in the previous assessments. More detailed descriptions are given in the appendices to this report. Most of the earlier work was carried out and reported in connection with the application for an operational license in 1987 (SSR, 1987). After that some complementary analyses were made and reported in a deepened safety report in 1991 (FSA, 1991).

In the previous analyses, the present situation with SFR-1 located beneath the bottom of the sea was defined as the marine or saltwater period. Land rise will cause a regression of the shoreline. The time period after the shoreline passes the repository was defined as the inland period.

3.1.1 Scenario methodology

As a part of the deepened safety analysis of SFR-1 (FSA, 1991), a methodology to systemise and visualise all Features, Events and Processes (FEP) that can influence the performance of the repository in the future was applied.

Influenced by the methodology outlined in a joint SKI/SKB scenario development project (Andersson et al., 1989) an initial FEP list was compiled, containing phenomena that potentially could influence the long term performance of the repository. From this list, FEPs judged to belong to the Process System (i.e. FEPs required for description of the performance of the system components and radionuclide behaviour in the repository system) were selected. The Process System was graphically displayed in a reversed event-tree structure, showing how FEPs are linked according to cause and effects.

The graphical description of the Process System was used as a base for a written description of the system in terms of transport pathways for radionuclides and initial state and evolutionary processes in the different barriers. In addition, a screening out of phenomena was made and motivations for judging the out-screened phenomena to have negligible consequences for the performance of the system were given in text.

The graphical and written description of the components of the Process System and their interrelations were used as a base for describing the radionuclide release from the repository for different scenarios. Two different types of scenarios were considered, a reference scenario and some extreme scenarios. The reference scenario was intended to describe the most realistic evolution of the Process System with land uplift as the scenario initiating FEP imposed on the Process System. However, conservatisms were applied when uncertainties in the understanding of the long-term performance appeared. The extreme scenarios considered phenomena that were assessed to be very unlikely, but could have large impact on dose to man if occurring.

3.1.2 Waste, radionuclide inventory and near field processes

Waste volumes and radionuclide inventory

The waste volumes and the radionuclide inventory assumed in the previous safety analyses (SSR, 1987 and FSA, 1991) were based on plans and prognoses made in 1987. These plans and prognoses in turn originate in plans and prognoses made in 1981.

The distribution of waste between the different repository parts was made according to the main principle that waste arising from cleaning of the primary water systems at the power plants and from the fuel storage pools are placed in the Silo while the remaining waste are placed in the caverns. In the deepened safety analysis (FSA, 1991) it was noted that some minor changes in the distribution of waste between the repository parts had been made since the finalisation of the final safety report (SSR, 1987). However, these changes were not included in the deepened safety analysis since it was concluded that the long-term safety of the repository would not be affected.

The radionuclide inventory in the different repository parts was derived from the assumed distribution of the waste. At the time when the deepened safety analysis was carried out (FSA, 1991) more recent prognoses indicated smaller waste volumes than previously assumed. However, these changes in waste volumes were not considered and the radionuclide inventory was conservatively assumed to be the same as in the final safety report (SSR, 1987). It was assessed that the activity content of ^{60}Co, ^{137}Cs and transuranics were overestimated by a factor of 3, 5 and 10, respectively (FSA, 1991).

The radionuclide inventory assumed for the different repository parts in the previous analyses is given in Table 3-1 in Appendix A1. For each repository part it was assumed that the radionuclides were evenly distributed in the waste.

Complexing agents and degradation products

Complexing agents may affect the solubility and sorption of radionuclides and thereby their mobility and release from the repository. In the previous safety assessments of SFR no credit has been taken to potential limits in radionuclide solubility so the focus has been on the effects of complexing agents on sorption. Organic complexing agents, such as EDTA, are used in decontamination processes and are therefore present in small amounts in the waste. Complexing agents may also be formed as a result of different degradation processes in the repository.

Experiments with low concentrations of EDTA and similar complexing agents showed an initial effect on sorption, but after a few months the K_d-values were similar to those obtained without complexing agents. Based on these results and the small amounts of complexing agents in the waste the effects on radionuclide sorption were neglected both in the final safety report (SSR, 1987) and in the deepened safety analysis (FSA, 1991).

Experimental results, available at the time of the deepened safety analysis (FSA, 1991), indicated that the degradation product from alkaline degradation of cellulose, isosaccarinic acid (ISA), could form strong complexes with tri- and tetravalent radionuclides thereby affecting the sorption of these nuclides. The effect on sorption was found to be negligible at cellulose to cement ratios below about 1 %. Since the estimated cellulose to cement ratio in BMA was found to be of the order of 1 %, the effect of ISA on sorption in BMA was considered in the deepened safety analysis (FSA, 1991). This was done by reducing the K_d-values by a factor of 50 for the trivalent radionuclides americium and curium and the tetravalent radionuclides plutonium and technetium.

Other potential sources for organic complexing agents are alkaline degradation of ion-exchange resins and bitumen. Based on experimental data and results from investigation of a natural occurrence of bitumen in an alkaline environment the effects of degradation products from ion-exchange resins and bitumen on sorption was assessed to be negligible in the deepened safety analysis (FSA, 1991).

Degradation processes

Some processes that can cause chemical degradation of *the bentonite and sand/bentonite barriers* in the Silo were addressed in both the final safety report (SSR, 1987) and the deepened safety analysis (FSA, 1991). These processes are:

- The transformation of the sodium bentonite to calcium bentonite with a higher hydraulic conductivity due to supply of calcium ions from the concrete.
- The formation of secondary clay minerals with poorer swelling properties than bentonite due to attack by hydroxide ions that are dissolved from the concrete.
- The formation of precipitates of calcite and brucite due to chemical reactions when calcium and hydroxide, dissolved from the concrete meet carbonate and magnesium from either groundwater or dissolved from the bentonite barriers. The percepitates could reduce the porosity of the bentonite barriers but, on the other hand, also make the barriers less swelling and more sensitive to fracture formation.

The effects of these processes on the properties of the bentonite barriers were considered by making a conservative selection of assumptions and data in the modelling of radionuclide migration in the bentonite barriers.

A number of chemical and physical processes that can affect the barrier properties of the *concrete materials* in the repository parts were addressed in the previous safety analyses. The extent of the different processes was assessed and processes judged to be of potential importance were studied in more detail. The chemical processes considered were:

- Leaching of alkali-hydroxides, portlandite and calcium-silicate-hydrate which is affecting the pH and ionic strength of the water as well as the porosity of the concrete materials
- Ettringite formation that will reduce the porosity of the concrete materials or cause fracture formation if the porosity in the materials and other accessible voids in the system not are large enough for the volume expansion associated with ettringite formation.
- Chloride intrusion and formation of Friedel´s salt that would have similar effects as the formation of ettringite.

- Precipitation of calcite and brucite which could reduce the porosity of the concrete materials.
- Recrystallisation of the calcium-silicate-hydrate gel causing changes in the pore- and crystal structure of the gel. This could affect the diffusivity and sorption properties of the concrete and also the hydraulic conductivity of the cement gel.
- Alkali-silica reactions, where alkali-hydroxides from the cement reacts with silica in the ballast material to form volume expanding products. This could reduce the porosity of the materials or create fractures in the materials if the accessible porosity not is large enough.
- Formation of calcium-silicate gel (puzzolan reaction), where calcium hydroxide from the cement reacts with silica in the ballast material. This could reduce the porosity of the concrete materials.
- Corrosion of reinforcement that could cause fracture formation in the concrete due to formation of volume expanding corrosion products.

The main physical process considered was fracture formation. The potential causes of fracture formation addressed, in addition to the chemical processes listed above, were:

- Settlement of the repository structures
- Mechanical impact on the concrete structures in the Silo due to uneven swelling of the bentonite barriers
- Movements in the rock and rock fallout
- Mechanical impact on the concrete materials in the Silo due to water uptake and volume expansion of ion-exchange resins.
- Build-up of high gas pressures in the interior of the Silo due to restricted gas release.

The effects of leaching on the water composition were considered in the selection of sorption data for the concrete materials in the Silo and the vaults.

The time for chemical depletion of the materials due to leaching was estimated to be long and the effects on the porosity small compared to mechanical impact and fracture formation that could occur earlier in time and have larger effects on the barrier properties. The presence of fractures in the concrete materials was considered in the radionuclide migration calculations in the selection of parameter values, e.g. hydraulic conductivity, and by making different calculation cases where the number, locations and sizes of potential fractures in the Silo structures were varied.

The potential reduction in porosity of the concrete materials due to e.g. precipitation of calcite and/or brucite was conservatively neglected in most of the radionuclide migration calculations. However, in the quantitative estimates of gas release from the Silo some calculation cases assumed that pores in the concrete or in the sand in the gas release devices were blocked by precipitates (see below). This was done despite that quantitative estimates of the extent and potential locations of such precipitates indicated that these would more likely be formed in the sand layer in between the concrete lid and the sand/bentonite above than in the concrete or in the gas release devices.

Gas generation and transport

Gas can be generated in all parts of the repository because of anaerobic corrosion of steel, microbial degradation of organic material and radiolysis. Using data available in the literature at the time of the previous safety analyses, estimates of the gas generation rates showed that the dominating contribution arises from anaerobic corrosion in all repository parts except in BLA where microbial degradation also contributes (SSR 1987).

In the previous analyses gas generation in the repository vaults was assessed to have no impact on the radionuclide release since the basic assumption was that there is no resistance to gas escape because of initial fractures in the surrounding barriers or lack of surrounding barriers. This means that no significant gas pressure build-up is expected and any water displacement that may occur by the gas is negligible compared to the hydraulic gradient driven water movement in the vaults. The Silo, however, is surrounded by bentonite and sand/bentonite that acts as barriers for both gas and water flow. This means that gas generated in the Silo interior has to create a certain over-pressure before it can escape from the Silo and that the water displaced by gas may affect the radionuclide release from the Silo.

As a part of the deepened safety analyses gas generation and gas transport from the Silo was modelled to estimate the impact on the radionuclide release from the Silo (FSA, 1991). In the base case it was assumed that the Silo is water-saturated initially and that the gas source is anaerobic corrosion of steel. It was further assumed that the generated gas would create gas channels through the porous concrete in the Silo in order to reach the gas release devices in the top of the Silo. For further gas escape through the top bed of sand/bentonite a gas pressure gradient has to be established over the sand/bentonite. Both the creation of gas channels in the Silo interior and the accumulation of gas and gas pressure build up beneath the sand/bentonite will expel water from the interior of the Silo.

The effect of restricted gas transport in the Silo interior, e.g. due to precipitation of calcite and brucite, and of fractures in the concrete floor and walls on the displacement of water from the Silo was studied in a number of variation cases.

3.1.3 Hydrogeology

Modelling

The main hydrogeological modelling was made as a part of the safety analysis reported in 1987 (SSR, 1987). Modelling was done in three dimensions, but with rather coarse meshes. The rock was described as a porous medium, and fracture zones were modelled by assigning the decided hydraulic conductivities to the finite elements representing the structures.

The simulations were carried out in three scales and a nested modelling approach was applied, such that the large scale models provided input to the smaller scale models. The simulations with the regional model (~ 85 km^2) were used to describe the influence of regional lineaments on the groundwater conditions in SFR-1. It also provided boundary conditions for the local model.

The simulations with the local model (~ 2 km^2) were used to describe the effect of local lineaments on the groundwater conditions around SFR. In the local model the Singö zone represented the western boundary. The horizontal zone H2 was extended to this boundary in the model, but a variation case with a less extensive H2 zone was also included in the analysis. The access and construction tunnels were modelled as a single pipe. The repository vaults were modelled as a single 10 m thick horizontal slab and the Silo as a rectangular block.

The repository component model (~ 1 km^2) was used to estimate the inflow to the different parts of the repository during the operational phase and the resultant potentiometric head field of the surrounding rock under present day drainage stage.

Marine period, land rise and inland period

The regional and the local model analysed the marine period (saltwater period) and the inland period as two different steady-state simulation cases.

For the marine period an "excess head" of 0.5 m was applied as a boundary condition at the Singö zone in the regional model. This was motivated by an excess head interpreted from extrapolations of measurements at the H2 zone that were made before construction of SFR-1. This boundary condition transmits into the local model and is the only driving force for groundwater flow in the simulations of the marine period.

For the inland period boundary conditions were altered in order to represent the effect of land rise.

In preparation for the deepened safety analysis (FSA, 1991) some aspects of land rise and its effects on the dilution in a domestic well was analysed. These calculations were carried out in two-dimensional vertical cross sections through the repository and along fracture zones downstream the repository using a semi-analytical approach. The calculations provided a qualitative picture of the transition from the marine period to the inland period. It was concluded that the location of the salt/fresh water interface is transient. During the marine period the groundwater flow through the repository is quite minor. Once the shoreline passes the repository the groundwater through flow will commence, but the magnitude and direction of flow will change as a result of the transient character of the flow. The time to reach pseudo steady-state conditions (provided that there are no other changes in boundary conditions) would be several thousands of years.

This latter modelling confirmed the earlier local model results what concerns the magnitude of the water flow in the repository rock after 1000 years when the shoreline passes the repository. Due to the continuation of the land rise the model predicted a further increase in the flow magnitude with a factor of 2.5 - 3.5 after 2500 years.

Wells and well-scenarios

The semi-analytical two-dimensional model was used to estimate the dilution in wells located in fracture zones downstream the repository. The fracture zones included in the model are the vertical zones 3 and 8 and the model also considers the horizontal zone H2 below the repository. In the calculations, no considerations were taken to the channelling character of flow in fractured rock or sorption, matrix diffusion or dispersion phenomena. Furthermore, the model is conservative since only limited account is taken to the contribution of flow from the rock mass surrounding the vertical cross-sections. In addition, the only source to the flow in zone H2 is water that has passed through the repository thereby neglecting the "clean" groundwater recharging above zone H2 and the contribution of shallow groundwater from the surrounding rock mass.

Simulation with the model for conditions prevailing after 2500 years showed a dilution of a factor of 3 in a well that is located 1 km from the repository. An assessment of the conservatism's introduced in the calculation lead to the conclusion that the dilution factor in the well should be at least a factor of 10.

Input to the radionuclide migration calculations

The flow through the repository vaults that was assumed in the radionuclide migration calculations was estimated from the water turnover in the slab representing the repository vaults in the local scale model. The model resolution did not allow for a direct division of the flow through the slab between the individual vaults. Instead it was assumed that the flow in each of the vault is in proportion to the bottom area of the vault.

The input to the Silo calculations was taken directly from the local scale model in terms of the Darcy velocity and water flow.

Parameters were provided both for the marine period and the inland period. In the radionuclide migration calculations carried out in the deepened safety analysis (FSA, 1991) the increase in water flow after the onset of the inland period (after 1000 years) was derived from the results of the semi-analytical two-dimensional modelling (see above).

3.1.4 Radionuclide transport

In the final safety report (SSR, 1987) radionuclide release calculations for each repository part were reported for the marine period and the inland period. In the deepened safety analysis (FSA, 1991) new calculations on the release from the Silo during the marine period were carried out. The main reason for updating these release calculations were that the sorption data had been updated and that the authorities in their review of the final safety report has asked for a more detailed analysis of the effect of gas generation on the radionuclide release from the Silo. The release calculations for the repository vaults were not updated. The deepened safety analysis therefore reported the same calculations as in the final safety report despite that they were carried out with an older set of sorption data. However, the effect of the differences in sorption data was estimated and reported in the deepened safety analysis report (FSA, 1991) considering also potential effects of the formation of complexes with isosaccarinic acid (ISA).

For the inland period new calculations were carried out for all repository parts in the deepened safety analysis. This was done in order to meet the authorities request on a more detailed analysis of the consequences during the inland period. Special attention was paid to the formation of ISA, gas generation in the Silo during the inland period, changes in the groundwater flow due to land rise and the possibility of different wells being the primary recipient for released radionuclides.

The base scenario calculations reported in the deepened safety analysis (FSA, 1991) are briefly summarised below (see also Appendices A3 and A5). To illustrate the effect of some possible, but not very likely situations some extreme cases were defined and analysed in the deepened safety analysis (FSA, 1991). These extreme cases are also briefly summarised below (see also Appendices A3 and A5).

General assumptions

The calculations were based on a number of pessimistic and simplified assumptions. Some general assumptions used in the calculations were:

- All activity intended for each of the repository parts is located in one Silo, one BMA vault, one BTF vault and one BLA vault.
- The activity within each repository part is evenly distributed in the waste.
- The repository is water saturated initially after closure and all radionuclides are immediately dissolved.
- During the marine period the groundwater flow is constant and directed upwards. At the onset of the inland period, 1000 years after repository closure, the groundwater flow momentarily increases a factor of 10 and becomes downward directed. From 1000 years to 2500 years after repository closure the groundwater flow increases linearly by a factor of 3.5.
- The water flow in each of the repository vaults is in proportion to the bottom area of the vault, see 3.1.3.
- At the onset of the inland period, the radionuclide inventory is compensated for radioactive decay during 1000 years, but not for any radionuclide release during the marine period.

Base cases, marine period

During the marine period the repository is located below the bottom of the sea. The hydraulic gradient is small and the groundwater flow is horizontal or weakly upward directed. The primary recipient is the sea located above the repository. The distance from the repository to the recipient is short and any retardation of radionuclides in the rock between the repository and the sea was therefore neglected.

Silo

The system inside the concrete walls, lid and bottom was modelled as a well-stirred tank. The hydraulic resistance of the bentonite barriers in the Silo was assumed to be large and transport by groundwater flowing through the Silo was neglected compared to transport by diffusion. Sorption in concrete and in the bentonite barriers was taken into account. Displacement of water and dissolved radionuclides from the Silo interior by gas flow and gas pressure build-up was also considered.

BMA

The calculations reported in the deepened safety analysis (FSA, 1991) were the same as those already reported in the final safety report (SSR, 1987).

It was assumed that the water flow through the vault is upward directed and evenly distributed and that the magnitude is determined by the groundwater flow in the rock. The system inside the concrete bottom, walls and lid was modelled as a well-stirred tank. The radionuclide release to the rock was modelled as transport by groundwater flow through the concrete lid and transport by diffusion through the concrete walls and mixing of the released radionuclides in the vault void above the concrete lid. Sorption in concrete was taken into account.

BTF

The calculations reported in the deepened safety analysis (FSA, 1991) were the same as those already reported in the final safety report (SSR, 1987).

It was assumed that the groundwater flow through the vault is upward directed and evenly distributed and that the magnitude is determined by the groundwater flow in the rock. The interior of each concrete tank in the vault was modelled as a well-stirred tank and radionuclides are released by diffusion and by groundwater flowing through the tanks. Sorption in concrete was taken into account.

BLA

It was assumed that the groundwater flow through the vault is upward directed and evenly distributed and that the magnitude is determined by the groundwater flow in the rock. The whole vault was modelled as a well-stirred tank where radionuclides are released to the surrounding rock with the groundwater flow without any sorption on the materials in the vault.

Base cases, inland period

In the radionuclide calculations it was assumed that the water flow through the different repository parts is the same as in the surrounding rock, and that the flow rate increases linearly by a factor of 3.5 during the time period 1000 to 2500 years after repository closure.

Silo

It was assumed that the bentonite barriers and concrete structures no longer could prevent water flow through the Silo due to chemical and mechanical degradation during the marine period. Conservatively it was assumed that the water flow through the Silo is the same as in the surrounding rock.

The system inside the concrete bottom, walls and lid was modelled as a well-stirred tank. The radionuclide release was assumed to occur through the bottom by groundwater flowing downwards through the Silo and by diffusion through the bottom and the walls. Sorption in the concrete walls and bottom and in the bentonite barriers was accounted for. The effect of neglecting sorption in the concrete bottom was also studied.

BMA

Two different model approaches were used in the release calculations, one where the transport resistance in the concrete structures and the system inside the structures were considered and one where this transport resistance was neglected. In the first model it was assumed that the water flow through the vault is horizontal and that a fraction of the total flow passes through the concrete structures. Radionuclides are transported through the concrete structures and the system inside by this flow and by diffusion.

In the second model the entire vault was modelled as a well-stirred tank and radionuclides are released from the vault by the groundwater flowing through the vault.

In both modelling approaches, sorption in concrete was taken into account. The first model was also used to study the effect of complexing agents formed by alkaline degradation of cellulose. This was done by lowering the K_d-value for some of the radionuclides.

BTF

The entire vault was modelled as a well-stirred tank where radionuclides are released from the vault by the groundwater flowing through the vault. Sorption in concrete was taken into account.

BLA

The entire vault was modelled as a well-stirred tank where radionuclides are released from the vault by the groundwater flowing through the vault. Any sorption on materials in the vault was neglected.

Radionuclide migration in the geosphere

In the previous safety analyses it was assumed that the primary recipient in the biosphere is a lake or a well located approximately 1 km downstream of the repository. The retention of the radionuclides in the geosphere for Pu-239 and Pu-240 for a transport distance of 1 km and the estimated groundwater flow 2500 years after repository closure was estimated from the results obtained by calculations using a channelling model. Sorption data for Pu used in the calculations were not considering effects of complexing agents or colloids. The retention of other radionuclides in the geosphere was not estimated in the previous analyses.

Extreme cases

To evaluate potential disturbances in the performance of the Silo some extreme calculation cases were analysed in the deepened safety analysis (FSA, 1991). In addition, the consequences of drilling wells into the repository vaults were analysed.

Silo

The extreme cases analysed were based on pessimistic assumptions concerning gas transport in the Silo interior and fracture formation in the concrete structures of the Silo. The consequences of these assumptions have implications on the volume of water that is expelled from the Silo and how far out into the Silo barriers the water is expelled and thereby on the release of radionuclides dissolved in the expelled water.

Drilling wells into the repository vaults

The consequences of drilling walls into the repository vaults were estimated by calculating the radionuclide concentration in the vaults at a time where the salinity of the water in the vaults are 10 times lower than during the marine period. The dose to individuals was calculated from the estimated radionuclide concentration in the vaults, the annual consumption of water and dose conversion factors for ingestion according to ICRP.

3.1.5 Biosphere and dose

A biosphere model for radionuclide transfer in the biosphere and resulting doses was set up as a part of the safety analysis reported in 1987 (SSR, 1987). The same model was used also in the deepened safety analysis (FSA, 1991) when updating the calculations for the marine period. However, for the inland period a simplified model was used where the only exposure pathway considered was via drinking water from a well.

Model structure

The biosphere was modelled by compartment theory, where the biosphere components were divided into physical areas with uniform properties. The exchange of nuclides between those compartments was described by rate constants expressed in turnover per year, also taking into account radioactive decay.

The main purposes with the analysis were to assess the radiological consequences to a critical group and population from potential releases from the repository. Therefor the model set up was divided into four spatial scales to be able to simulate the turnover of the radionuclides from the local to the global area.

The local zone was used for the calculation of individual doses to a critical group living around the entrance of nuclides to the biosphere. For releases during the first thousands years, the marine period, the local zone consisted of a part of Öregrundsgrepen. After land rise, the inland period, the zone consisted of a well and a lake area.

In **The regional zone,** the dispersion of radionuclides in an expanded area around the point of discharge was considered. The recipients for discharged radionuclides were the entire Öregrundsgrepen during the marine period and the lake during the inland period. The local zones were consequently included in the regional zone.

The intermediate zone consisted of the entire Baltic Sea.

The global zone included the entire world.

Exposure pathways and dose calculations

In the assessments only internal exposure pathways were considered. The internal exposure pathways were those connected to agricultural practices or consumption of marine products.

The exposure to a critical group in the local zone was assumed to occur via consumption of fish during the marine period. No direct intake of water was considered since the water in Öregrundsgrepen is brackish. For the inland period it was assumed that the exposure to a critical group occurred simultaneously from the lake and well water. The water in the lake is used for irrigation and transfer of radionuclides to man occurs via agricultural products and also via fish from the lake. The water in the well is used as drinking water for both humans and animals and the exposure pathways considered are directly via intake of well water and indirectly via consumption of meet and milk.

The regional-, intermediate- and global zones were considered in the calculations of collective dose. The exposure pathways considered for the regional zone were similar to those considered for the local zone, excluding exposure of the regional population through the local area of the discharge point in Öregrundsgrepen during the marine period and through well water during the inland period. Potential values of annual yield for the agricultural area and Öregrundsgrepen were used in the calculations.

The exposure pathways considered for the intermediate zone were via consumption of fish and algae products. Algae were considered because they may become an important protein source in the future. These exposure pathways were also considered for the global zone together with terrestrial exposure pathways.

3.2 Regulatory Reviews

This section summarises the main comments made in the regulatory reviews connected to the evaluation of the operating permit (SKI, 1988 and SSI, 1988) and in the joint SKI/SSI review of the deepened safety assessment (SKI/SSI, 1992). Only comments of interest to increase the quality of the renewed safety assessment are included here. The appendices (A1 to A6) contain a more thorough listing of the comments made in these reviews.

3.2.1 Scenario methodology

In the 1992 review SKI and SSI found the scenario work to be satisfactorily carried out in that the most important scenarios from a safety point of view were identified and described. However, some criticisms were given to the coupling between the scenario work and the selection of calculation cases.

The reviewers concluded that describing the performance of the repository as a reversed tree of events becomes complicated and extensive if it includes all potential phenomena affecting the future evolution of the repository system. Therefore, simplifications and conservative assumptions were made at all levels from the initial identification of processes and their interrelations to the quantitative analysis of the calculation cases. The view of the reviewers is that with this approach the selection of calculation cases could have been made more complete, i.e. alternative calculation cases could have been selected and analysed. It was also pointed out that making conservative assumptions and choosing conservative models and parameter values throughout the whole analysis might result in a too conservative and totally unrealistic description of the repository performance. Therefore, if possible, a reference case or a base case should be based on more reliable and realistic assumptions, and the uncertainties could be analysed by using alternative models and making variations in parameter values.

3.2.2 Waste, radionuclide inventory and near-field processes

Distribution of radionuclides in the repository

The reviewers noted that the distribution of the radioactivity between the Silo and other repository parts is changing. In the final safety report 92% of the radioactivity is to be found in the Silo, but based on prognoses from 1992 the radioactivity has decreased to 87%. Changes in waste allocation can lead to changed prerequisites for the calculations.

Complexing agents and degradation products

Complexing agents in the repository can increase the mobility of radionuclides and thereby increase the radionuclide release from the repository. The reviewers requested control of the amount of complexing agents in the deposited waste and research efforts on possible formation of complexing agents by degradation of material in the repository. In addition, continuos updating of these areas are requested. The degradation of organic material in the waste and the formation of complexing agents (e.g. ISA) may also lead to decreased sorption.

In the deepened safety assessment, for BMA, the sorption constants for plutonium, americium and technetium were reduced by a factor of 50 to consider the formation of complexes in the presence of a substantial amount of cellulose. The reviewer considered that the determination of the factor 50 was not clear and could be improved. According to the reviewer, the reduction factor for BMA could be as large as 75 - 300. For unfavourable cases this factor may be 3 000. Cellulose in the Silo and BTF cannot be neglected. The reviewer comes to the conclusion that the assumed reduction in K_d for influenced radionuclides in BMA and the uncertainty involved in the evaluation of available experiment needs to be studied in more detail. According to the reviewer, similar factors could be used for sorption in bentonite and sand/bentonite.

Other waste related topics

In addition, the following waste related topics have been noted in the reviews to be potentially important:

- importance of controlling materials and substances in the waste that may influence the long-term performance,
- microbial degradation of organic material in the waste and related gas formation,
- sulphate and sulphate sources in the waste influencing ettringite formation.

Degradation processes

According to the reviewer there are large uncertainties in estimates of the mechanical properties of concrete in structures after 500 - 1 000 years due to formation of ettringite. It is possible that the concrete walls will be damaged to such an extent that the concrete walls do not impede water flow. However, only marginal influence is expected on the radionuclide release.

The maximum gas pressure the Silo can withstand is estimated to be 280 kPa. According to the review this figure can be too low for the fresh concrete and a higher value is expected for the first part of the marine (saltwater) period. The implication is that a potentially higher gas pressure can be built up resulting in an increased amount of displaced contaminated water.

The formation of ettringite in concrete and the water uptake in bitumen conditioned ion-exchange resins can lead to swelling and pressure build-up. The reviewers say that for longer time periods, fracture formation of waste packages and concrete structures must be considered.

Gas transport

The reviewers considered that the assessment does not show that the ability of the porous concrete to carry off the gas from the Silo will be retained for long times. However, the consequences of a variation in the transport properties of the porous concrete have a small significance

3.2.3 Hydrogeology

Modelling

Both in the 1988 and the 1992 review SKI lacked an in-depth discussion of the uncertainties in the hydrogeological properties such as possibilities of additional fracture zones outside the vicinity of the rock tunnels and the spatial variability of the hydraulic conductivity in the fracture zones and rock mass. The values of permeabilities used in the assessment show a large uncertainty, they could be five and ten times larger for fracture zones and mass rock respectively. However, these permeabilities could also be smaller than the values used in the assessment. Variation cases set up by the reviewers assumed uncertainties of one order of magnitude. The reviewers also suggested that the existence of possible sub-horizontal fracture zones should be investigated.

The reviewers identified that the performed estimates of the groundwater flow through the caverns probably are of the right order of magnitude, but based on very crude modelling. In the 1988 review, SKI noted inconsistencies in the mass balance evaluation over the elements representing the repository. Much, but not all, of the mass balance inconsistencies were resolved by complementary groundwater flow modelling carried out by SKB in 1990 (Ström, 1990). The remaining inconsistencies are due to coarse finite element meshes necessary for the computer capacities available at the time of simulation.

Marine period and land rise

The interaction between land rise, saltwater and groundwater movement is only qualitatively described in the assessment. It was noted that the transient effect from the changing boundary conditions needs to be incorporated if a quantitative study is undertaken.

The reviewers suggest that the assumed excess head during the marine period may be incorrect. It may be more probable to assume almost zero flow during the marine period.

The inland case

The groundwater flow calculations indicate that most groundwater flow through the rock vaults occur through the floor. However, it was noted that the accuracy of these old calculations does not really support such precision and it was suggested that for the inland case the flow is probably horizontal to downward.

SKI's calculations indicate that for the inland case the groundwater flow, including the flow through the vaults, may increase by a factor of 4 if the sediment layer on top is removed. Uncertainty regarding the water flow direction is also pointed out.

Wells and well-scenarios

In the 1992 review SKI and SSI noted that the inventory of wells in the Forsmark area suggests that the probability of finding a well in the area will be relatively high. Regarding the inland period, the reviewer's opinion was that a well drilled earlier and nearer to the repository should be studied in order to get a complete description of the well scenario.

The reviewer commented that the estimated dilution factor for the wells is based on 2-D modelling and therefore does not take credit for the dispersion transverse to the flow-direction that could be substantial. They also pointed out that a dilution factor of 10 is a careful choice, but that wells closer than 1000 m from repository are also a possibility and these wells should have a lower dilution factor. For example, for a well located directly above SFR, where the assessment uses a dilution factor of 2, the reviewers proposed no dilution.

3.2.4 Radionuclide transport

Generally, SKI and SSI in the 1992 review suggested that the use of simple models with pessimistic values for the few parameters included in the models, may result in too unrealistic description of the system. They also indicated that the extreme cases accounted for are unrealistic and improbable. On the other hand, the reviewers also noted that SKB has avoided combining extreme cases judged to be too improbable. The reviewers considered that combination of different not too pessimistic cases would be illustrative. This may be done in a probabilistic way.

As already discussed above, the reviewers stressed that the estimates of the water flow through the repository is uncertain. Higher values than used in the SKB safety assessments could not be excluded for the inland case.

Modelling each part of the repository as a well-mixed tank leads to an overestimate of the source term. The reviewers stated that the release would be significantly reduced if water only flows through certain channels in the waste containers. If the distance between these paths is large, the radionuclide transport is controlled by diffusion. The reviewers noted that the assumption of a well-stirred tank used for the Silo means that many difficult problems are avoided (e.g., the course of the concrete degradation or preferential paths through the Silo). This means, however, that the consequences are over-estimated. On the other hand, regarding BMA the reviewers are critical to the source term model based on assessment of the water flow through the concrete building. If the whole cavern is modelled as a well-mixed tank, the doses could be one order of magnitude larger.

According to the reviewers, transport properties of the geosphere are missing. Formation of colloids and the existence of colloid-forming substances in the repository are factors that increase the uncertainties. Radionuclide complexes formed inside the repository in the high pH environment may be destroyed by the lower pH existing outside the repository. The so formed free radionuclides may then be sorbed on the fracture surfaces and rock matrix.

In the joint SKI/SSI review in 1992 it was also suggested that:

- uncertainties in the models (mathematical, physical or geometrical) can be addressed by alternative models,
- parameter uncertainties may be studied by sensitivity analysis or probabilistic calculations.

3.2.5 Biosphere and dose

In the 1988 review (SSI, 1988), SSI concluded that the assessments corresponded to the general status of environmental modelling. However, they identified that there was no connection between surface sea and global atmosphere. This underestimates the collective dose for ^{14}C in comparison to the calculations performed by SSI. This error in combination with the high importance of exposure from ^{14}C resulted in the initiation of a specific study of ^{14}C.

In the joint SKI/SSI review (SKI/SSI, 1992) the main comment was that the documentation and model descriptions should be improved as well as reasons for uncertainties be discussed. The reviewers also pointed out that realistic estimates of doses should be made. However, in their assessment, the reviewers focused considerably on the location of wells, as wells were deemed to give the highest exposure.

4 Operation and Incidents

The main focus of the final safety report (SSR, 1988) was on the long-term function of the repository. Nevertheless, an important part is the description of the operation of the repository and the analysis of possible incidents during operation.

4.1 Operation

The conclusion after going through the final safety report is that the description of the operation of the repository needs very few adjustments. No further work is needed in this area until the next version of the final safety report is to be written.

4.2 Incidents

A thorough analysis of possible incidents during operation was made in 1987 (Bjälvenlind et. al., 1987). The report lists a series of different incidents; failure of power supply, failure of ventilation, failure of drainage pumps, failure of fire pumps, transport incidents in the facility, incidents in emplacement of the waste, fire incidents and some extreme external cases like flooding or large fall-out of rocks.

Going through this report has shown that the analysis is accurate in general. But since the material is over ten years old and systems in SFR have been modified, a complete update should be made of the material.

5 Inventory and Waste Allocation

In this chapter, identified needs of improvements concerning the information on waste inventory and waste allocation are summarised. The waste inventory includes waste volumes and radionuclide inventory, but also type and amount of chemical substances in waste and packaging materials that may affect the performance of the repository. In addition, potential needs for updating the information on types and volumes of construction materials are addressed here. More details are given in Appendix A1.

5.1 Inventory

5.1.1 General

The radionuclide inventory used in previous assessments, which is almost identical with the limits in the licence for operation of SFR-1, will be used as the inventory in the safety assessment as a conservative case. All improvements suggested are aimed at deriving a more realistic radionuclide inventory for assessment of a realistic case. Changes in the conservative inventory will only be made if the update of the inventory indicates a need for raised limits, or if the radionuclide release calculations show that some limits must be decreased in order to make a safety case.

5.1.2 Waste volumes

The waste volumes are the foundation of the inventory. Good estimates of the volumes of the different waste types are therefore needed. The latest prognosis of the waste volumes was made in 1995. This prognosis differs in several ways from the earlier prognosis made in 1987 because the waste producers have changed their procedures for handling of the waste (see Appendix A1). In order to get the best possible estimate of the waste volumes for the new safety assessment it is proposed that a thorough prognosis of the future waste production should be made in 1999. Until then the latest one from 1995 could be used in the calculations.

5.1.3 Radionuclides

The amount of strong gamma emitting radionuclides like Co-60 and Cs-137 in a waste package are easy to measure, while the amount of alpha and beta emitting radionuclides in a waste package cannot be measured directly because of attenuation of radiation in the waste package materials. This means that a radionuclide inventory to a large extent is derived by the use of correlation factors between radionuclides that are difficult to measure and radionuclides that can be measured. It is suggested that the correlation factors used in the previous assessments of SFR-1 are reviewed and updated since new improved measurements now are available as well as more extended compilations of correlation factors that to some extent take the radionuclide "history" into consideration.

Some specific radionuclides of interest are identified and listed below (see Appendix A1).

^{14}C In the previous safety assessment organic ^{14}C was predicted to be the radionuclide that will dominate the individual doses after a few hundred years and the collective dose after 2000 years (the inland scenario). Since it is not possible to measure the inventory of ^{14}C in a waste package the inventory is derived by the use of a correlation factor to "easy to measure" radionuclides. The correlation factor used is very uncertain and it is therefore suggested that both the correlation factor and the previously assumed distribution between inorganic and organic ^{14}C is reviewed and updated.

^{36}Cl This radionuclide is very difficult to measure and was not included in the previous assessment. Although the authorities in their reviews of the safety reports say that the doses arising from chlorine probably will be small, the inventory should be estimated.

^{59}Ni This long-lived nuclide limits the close-to-the-core metal scrap that can be deposited in SFR-1. There is an on-going research project that aims to provide a better measuring method. This should make it possible to improve the knowledge about the ^{59}Ni inventory.

TRU, ^{90}Sr These radionuclides are subject to a special program. The database from this work should be used to calculate a better (more detailed) inventory. Calculations of the content of different plutonium nuclides in SFR-1 waste and estimates of the distribution of TRU- nuclides in the repository vaults should be performed.

5.1.4 Chemicals

Chemicals are here defined as chemically pure substances, for example salts, solvents and organic substances. Also mixtures of what is usually defined as "chemical-technical" products are included in the definition, e.g. detergents and concrete admixtures.

Chemical substances, both in the waste, the conditioning material, the waste packaging and in the construction materials are of great interest since even small amounts can seriously disturb the long term performance of SFR-1. For example, the presence of strong complexing agents like EDTA, citric acid and oxalic acid, which are used as decontamination solutions and detergents in the power plants, may affect the sorption of radionuclides. Another chemical of potential importance is sulphate, which can react with cement constituents and change the properties of concrete.

SKB has already started a project to make a better inventory of which chemicals are appearing in SFR-1. There is also a special program about concrete admixtures. These two projects will improve the knowledge of the chemical substance inventory of SFR-1 (Appendix A1).

5.1.5 Waste and construction materials

Waste materials here refer to different metals, concrete, plastic, wood and bitumen in the waste packages, i.e. materials both in the waste and in the waste packaging. The construction materials are here the concrete in the structures, the shotcrete in the vaults and tunnels, the backfill in the vaults and tunnels and the bentonite barriers in the Silo.

The construction of SFR-1 and the materials used are well known and there are no obvious needs for improvement, except possibly of the amount of reinforcement that can generate gas. The waste packaging and the matrix are also well defined in terms of which materials are present. The accuracy in the amounts of materials depends on how good the prognosis of the volumes is.

The waste composition in SFR-1 is based on information given in the waste type descriptions on the average composition of the waste. In that respect there seems to be no need for any update of the ion-exchange resins waste. However, recent updates on the composition of the scrap metals and refuse waste (Riggare, 1997) show that earlier estimates on average compositions are inaccurate. The uncertainties are still large in this area and a further look into this question should be beneficial.

5.2 Waste Allocation

Since the previous safety assessment there have been some changes in the allocation of some waste types to the Silo and the rock vaults. Waste types that in earlier prognoses were planned for the Silo are now placed in BMA, which means a higher radionuclide activity in BMA compared to the earlier prognoses. The waste allocated to BTF in the earlier prognoses was ion-exchange resin in concrete tanks, only. Since then new waste types have been approved for disposal and at least one of them containing ashes in double lid steel drums is now placed in one of the BTF vaults (1BTF). Such changes in waste allocation will affect the inventory of radionuclides, chemicals and other materials in the Silo and the repository vaults.

In the previous safety assessments it was assumed that the total estimated amount of radionuclides and other components in the waste destined for the Silo or the rock vaults is evenly distributed within each repository part without considering the actual positioning of different waste types. Today, when the information about the SFR waste is collected in a database, it should be possible to increase the resolution in the description of the waste to a waste type level. This will in turn allow for waste type specific considerations in the analyses of barrier evolution and radionuclide transport.

6 Scenario and System Analysis

In order to evaluate the performance of the SFR-1 it is necessary to develop a system description model, to identify scenario initiating events and conditions, to select quantitative evaluation models, to identify the information flow between models and to select quantitative calculation cases. This chapter outlines the proposed approach to these tasks. Appendix A2 provides further information.

6.1 Development of a System Description Model

6.1.1 Methodology

An important starting point in a safety assessment is to identify which processes and which interaction between processes that needs to be modelled. It is suggested to address this need by developing an interaction matrix of the SFR process system.

The basic principle of an interaction matrix is to list the parameters defining the properties and conditions in the physical components of the system studied along the leading diagonal elements of a square matrix. Events and processes that are influenced by and affects the properties and conditions defined in the leading diagonal elements of the matrix occur in the off-diagonal elements of the matrix.

An interaction matrix for the SFR-1 repository should cover the Silo and the repository rock vaults, the rock around and above the repository and the surface environment. Instead of compiling all information into one matrix it is proposed that three sub-matrices are constructed, one repository/near-field matrix, one geosphere matrix and one biosphere matrix.

The first step in developing the interaction matrices is to define the sub-system to be included in each matrix in terms of physical components, initial state of these components and boundaries between the subsystem as well as the external boundary and boundary conditions of the whole system. The parameters required to describe the properties and conditions in the physical components of the sub-systems are then listed in the leading diagonal elements of the matrices. After that, binary interactions between diagonal elements are identified by going through all off-diagonal elements in the matrices using a clock-wise convention for the direction of the interaction. Each interaction is documented by defining the interacting phenomenon as well as the parameters in the two interacting diagonal elements that are involved in the interaction.

The contents of the matrices are audited against different international lists of FEPs (**F**eatures, **E**vents **P**rocesses). Such lists can, of course, be used as checklists already during the identification phase. The important thing is that these lists are used in order to ensure that all relevant FEPs are considered in the matrices.

Finally, the importance of all identified interactions in the matrices is judged using a pre-defined priority scale. This is done for the initial and boundary conditions selected as a base/reference scenario, but also for other scenario initiating events and conditions.

The selection of scenario initiating events and conditions are described in 6.2. The result of the prioritisation should be properly documented in terms of assigned priority and motivation to the assigned priority.

6.1.2 Expected outcome

The suggested procedure would potentially lead to the following results:

- the interaction matrix provides a qualitative description of the evolution of the repository system and a documentation of all judgements made in developing the description,
- development of the interaction matrix is an efficient means of transferring knowledge on the system behaviour among the project team members and would thus be an important means of achieving integration and consistency in the quantitative analyses work,
- the interaction matrix can be used to judge the relevance of potentially important scenario initiating events and conditions,
- the interaction matrix can be used to check whether the quantitative models and model couplings considered actually address processes and interactions that are judged to be important on the qualitative analysis level.

6.2 Selection of Scenario Initiating Events and Conditions

The evolution of the SFR-1 will be influenced by future events (such as re-saturation, land rise, climate change, human actions etc.) and by the initial state of the repository at closure. Such events and conditions are called scenario initiators. Evidently, it is necessary to identify the scenario events and conditions that would have a significant impact on the analyses, and to motivate why others need not be analysed.

It is proposed that the project group do the final selection of scenarios. Going through existing FEP-lists may aid in this process. A base/reference scenario based on realistic initial and boundary conditions, i.e. considering expected initial states of the barriers and including land rise, is used as a starting point for developing other scenarios. Other scenarios may be identified after review of existing FEP-lists and after review of scenarios considered in the on-going safety assessment of the spent fuel repository, SR97. Based on the opinion of the regulators after reviewing the previous safety analysis of SFR drilling of wells near the repository should probably be considered either in the base/reference scenario or as an alternative scenario.

6.3 Model Selection and Identification of Information Flow Needs

The interaction matrix and the list of scenario initiating events and conditions do not provide quantitative information on the repository performance. There is a need to *select* *models* and to *identify the information flow* between the models selected. To facilitate this procedure it is proposed that a flow chart of the models and information between models is developed.

6.3.1 Development of a model information flow diagram

A model information flow diagram is a graphical illustration of the models and the information flow selected for quantitative analyses in the assessment. Figure 6-1 displays an example of such a diagram, building on a similar diagram developed for use within SR 97 (Andersson and Stigsson, 1998).

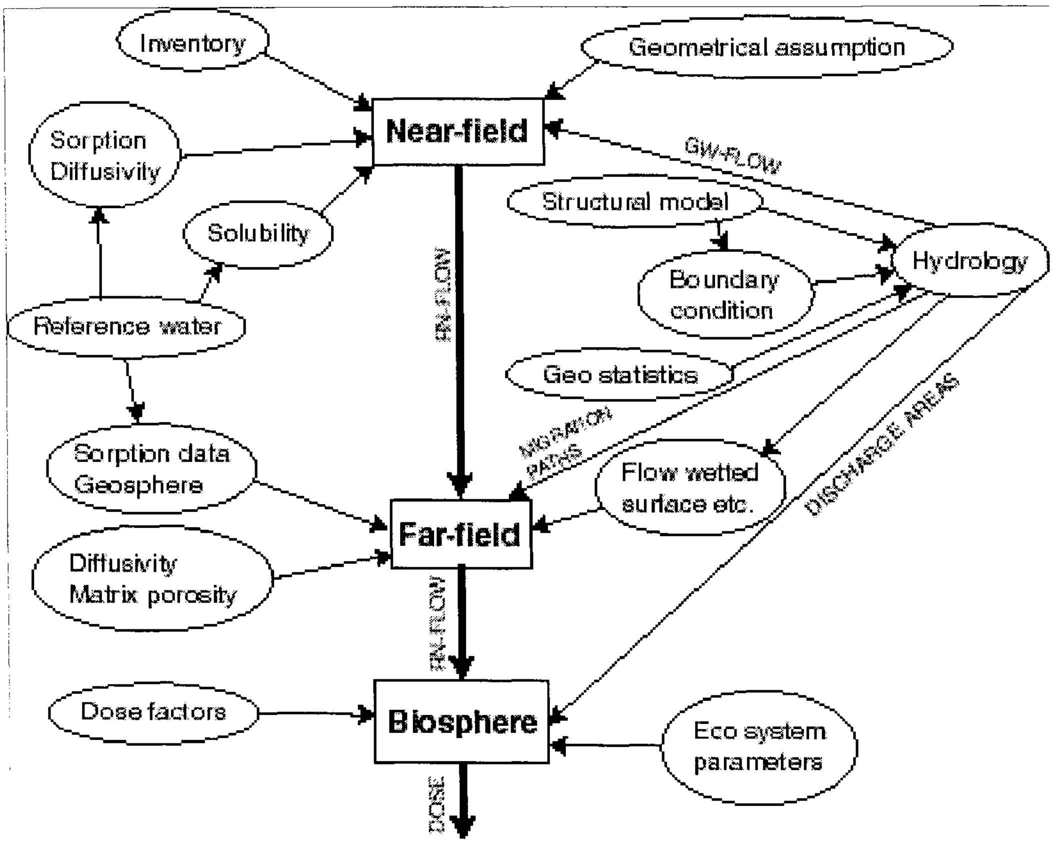

Figure 6-1. A rough sketch of a model information flow diagram for the SFR analyses. Further developments and adoptions to the actually planned analyses are needed.

A model information flow diagram could be used for:

- motivating the model selection by checking the consistency between models as well as information flow actually selected and the important interactions identified in the interaction matrices,
- providing a framework for assuring consistency in data transfer between different analyses and as a tool for planning of a logical order of analyses.

The following development steps are foreseen:

- The starting point should be the models and analyses already suggested for the coming analyses (see proceeding chapters and the appendices). These models and the information flow already identified will be sufficient to construct a first version of a model information flow diagram, building on the diagram displayed in Figure 6-1.
- The model selection (and the diagram) should be audited against the interaction matrix developed within the framework of the scenario exercises discussed previously.
- Results of scoping calculations may further be used for reviewing the diagram.
- There may be a need for different variants of the model chain (and diagram) in order to analyse different scenarios.

6.3.2 Quality control in assumptions and data transfer

Experiences with previous assessments indicate a great need for a quality control of assumptions made and data used in different supporting analyses. The model information flow diagrams will be an important tool for achieving such quality control, but additional efforts are necessary.

Issues to consider include:

- official data and data ”freezes”,
- who is authorised to change input data and when,
- procedures for collecting and distributing data/assumptions
- procedures for ”complaining” on data/assumptions (where to complain and how to take care of complaints),
- audits.

6.4 Development of Calculation Cases

There is a need to develop calculation cases for quantifying the scenarios selected. Calculation cases are developed for the actual model chain selected for the scenarios.

6.4.1 The calculation case selection task

The following needs to be considered:

- calculation case selection is a specific task that needs to be included in the project plan,
- calculation cases need to reflect actual model capabilities when describing scenarios and variability (e.g. one may need to represent time evolution in a specific scenario with a sequence of modelling steps).

6.4.2 Handling of data uncertainty and variability

It has been decided to concentrate the analysis on a *best estimate* case and a *conservative* case. This decision affects the development of underlying information.

A "best estimate" would ideally illustrate how the repository system would operate in "reality", but the term is judgmental and does not have a clear statistical meaning. The view is taken that the best estimate should represent models that are based on the present understanding of the phenomena, uses parameters derived from actually available data, and is consistent such that the model jointly can "explain" different data. The call for consistency implies a rigorous quality assessment of assumptions and of the used data and methods.

One way of bounding the impact of uncertainty is to construct a "conservative" case exploring the robustness of the system. The conservative case should only be formulated on the integrated analysis level and be based on an assessment of the uncertainties and correlations in the supporting analyses.

It is not suggested that a complete probabilistic assessment should be carried out. However, it is still necessary to acknowledge the underlying uncertainties and the *spatial variability* of waste composition and geosphere properties. Furthermore some of the supporting analyses may be of a probabilistic character.

7 Analysis of the Evolution of System Components

This chapter summarises already identified needs of new analysis of topics related to the evolution of properties and conditions in the near field (Appendices A3 and A5), far field (Appendices A4 and A5) and biosphere (Appendix A6). The work with the development of a system description model (see Chapter 6 and Appendix A2) may display additional needs not yet identified and may also show that some of the analyses proposed in the following sections are of less importance for the performance of the repository.

7.1 Near field

7.1.1 Hydrology

The water flow in the different repository parts is of importance for barrier alteration processes and for estimation of the radionuclide dissolution and release from the repository. In the former assessments the Silo and the caverns were not explicitly included in the hydrology models. Instead the water flow in the different repository parts was estimated from the water flow in the rock in the repository area without considering the hydraulic properties of the barrier materials in the different repository parts and interactions between these.

To be able to get a better description of the water flow in the different repository parts the need of an improved modelling of the near-field hydrology has been identified (Appendices A3 and A5). The important aspects that should be addressed are:

- The distribution of incoming water between the Silo and the different caverns during the saturation phase and in the long-term perspective considering the hydraulic properties of the barriers in the Silo and the caverns. The time and mode of saturation could be of importance for the Silo in terms of bentonite buffer behaviour and initiation of barrier alteration processes, e.g. corrosion, and radionuclide dissolution, see sections 7.1.4 and 8.1.
- The distribution of the water flow between the different barriers within the caverns considering the hydraulic properties of the different barriers. This is of importance since water mainly will flow in the high conductive parts of the caverns, which initially is the empty space or the gravel backfill outside the concrete buildings. This may change with time due to alteration processes affecting the hydraulic properties of the concrete, see section 7.1.4.
- The effect of plugs and location of plugs and tunnels and backfill in tunnels on the magnitude and direction of water flow in and between the Silo and caverns.
- The transient effects of land rise and a moving saline-fresh water boundary on the flow situation in the repository parts, see Appendix A4.

- Time when saline-fresh water boundary reaches the repository parts and water chemistry changes.

The implications on the groundwater modelling in order to account for these developments are discussed in section 7.2.2 and the model proposed for the near-field hydrology is described in Appendix A4. This model needs input on changes in hydraulic properties of the different barriers due to alteration processes, i.e. results of the analyses mentioned in the following sections.

7.1.2 Gas generation and release

Gas generation in the Silo and the release of this gas from the Silo may directly affect radionuclide release from the Silo by expulsion of water containing dissolved radionuclides. Accumulation of gas in the Silo interior may also indirectly affect the release of radionuclides if gas cannot escape without first creating high internal overpressures with cracking of the concrete barriers as a consequence. It is therefore proposed that gas generation rates and escape mechanisms are re-evaluated. The purpose should be to estimate potential effects of gas and gas pressure build-up on the physical properties of the different barriers in the Silo and in the caverns and possibly also on the distribution of water flow in the interior of the caverns and the Silo (Appendices A3 and A5).

The gas generation rates should be made as a function of time considering the following specific aspects:

- changes in waste composition and the use of alternative packaging in the Silo and the caverns,
- the time required for water saturation and initiation of anaerobic corrosion and gas generation in the Silo and maybe also in the caverns,
- the impact on the gas generation rate if corrosion starts at different times in different parts of the Silo interior,
- new information on corrosion rates.

An update of the modelling of the gas release is also proposed in terms of:

- A re-evaluation of the initial and potential changes in gas transport properties of the different materials in the Silo, such as internal voids and pore-size distributions in the porous concrete and concrete structures in the Silo. The actual composition of the porous concrete and information from the filling of the Silo should be utilised in this work.
- An update of the maximum internal design pressure of the Silo. In the previous analyses this maximum internal pressure was set to 280 kPa. If the Silo structures allows for a larger over-pressure than that, clogging of gas escape paths could result in a larger volume of water expelled by gas.
- An investigation of the need of a more detailed model of the gas release and pressure build-up in the Silo.

7.1.3 Water composition

The water composition in the Silo and in the caverns is of importance for chemical barrier alteration processes as well as for radionuclide solubility and sorption and, in some cases, for radionuclide diffusion. The main parameters of interest are pH, redox conditions, ionic strength (salinity) and concentration of complexing agents. The need for new analyses of the evolution of the chemical conditions has been identified and the specific aspects that should be addressed are summarised below (Appendices A3 and A5).

- The groundwater composition in SFR-1 should be defined since it affects the chemical conditions in the near field, at least during an initial period after repository closure. Information from the SFR control program on groundwater chemistry can be used for this purpose.
- The evolution of pH and ionic strength in the interior of the concrete buildings, but also outside these due to leaching of the concrete, should be estimated. Both pH and ionic strength are of importance for radionuclide sorption and the ionic strength can affect the diffusivity of some radionuclides. The dispersion of a pH plume could result in chemical alteration of gravel backfill and the near-field rock by dissolution of silica and precipitation of secondary CSH-phases, see section 7.1.4. In addition, there is a potential for colloid formation at the front of a pH plume.
- The "research" type work on alkaline degradation of cellulose that generates the strong complexing agent isosaccarinic acid, ISA, should be continued. The work should address issues like required pH for formation, yields, mechanisms and conditions for potential ISA degradation, sorption of ISA, mechanisms for forming radionuclide complexes, other potential degradation products that can form complexes, etc. The aim should be to derive a consistent and sound approach for quantification of the effects of ISA on radionuclide sorption and solubility.
- The effects on near-field water composition of a potential future intrusion of "fresh" water in SFR-1.

7.1.4 Barrier alteration processes affecting transport properties

The amount and characteristics of pores and fractures in the different barrier materials in the Silo and the caverns are of importance for the transport of water, gas and radionuclides in the different repository parts. In addition, the material composition of the barriers is affecting the radionuclide transport since it together with the water composition determines the sorption of the radionuclides in the barriers.

The main barriers of interest in the different repository parts are the waste matrices of bitumen or cement, the concrete packaging, backfilled concrete between the waste packages, concrete structures, bentonite and sand/bentonite, and potential filling materials outside the concrete buildings, e.g. sand or gravel. A number of new analyses have been suggested with the aim to obtain better estimates of the evolution of the transport properties of these barriers. The suggested analyses are summarised in this subsection (Appendices A3 and A5).

Steel packaging

Steel packagings are usually considered to be of minor importance as a barrier because it is difficult to show that they not are permeable to water already at repository closure and because corrosion anyhow will make them permeable within a short time period after repository closure. The corrosion products could potentially delay the transport of radionuclides by sorption and/or coprecipitation, but this effect is usually neglected. Thus, the main interest of corrosion is as a gas generation source.

In the Silo and in the BMA and BTF caverns where large amounts of concrete are present the effect of sorption onto steel corrosion products probably is of negligible importance. However, in BLA where the corrosion products are the only materials with a potential sorption capacity, except for the concrete floor in the cavern, potential sorption onto the corrosion products could affect the radionuclide release and this effect should maybe be included in a "realistic " calculation case.

Water uptake in bentonite and sand/bentonite barriers

The water uptake and development of swelling pressures in the bentonite buffer after repository closure should be studied with the purpose to estimate whether uneven swelling may take place and if this may have consequences for the integrity of the Silo structures. In doing this, information from the SFR control program on water uptake in bentonite could be utilised.

Water uptake in bitumen

The water uptake in bitumen conditioned waste will cause swelling of the waste and open up connected pores and fractures in the bitumen matrix. The properties of the bitumen matrix will then be affected by the extent of water uptake as a function of time. Extensive swelling of the bitumen matrix may also exert a load on surrounding barriers if not enough expansion volume is available. To be able to assess the impact of water uptake and swelling of bitumen-conditioned waste on the transport properties of the bitumen matrix itself and the surrounding barriers the water uptake process should be studied, see also section 8.1.

Chemical interaction between concrete and bentonite barriers

The chemical interactions between the concrete structures and the surrounding bentonite barriers may alter the properties of these. An updating of the former analyses utilising recent findings and research results has been suggested since it is believed that this may provide arguments for applying more realistic assumptions concerning the degree of alteration and changes in hydraulic properties of the concrete structures and bentonite barriers.

Chemical alteration of concrete

There are a number of chemical processes by which the transport properties of concrete may change with time. In former assessments the consequences of concrete leaching, formation of volume expanding monosulphate/ettringite and Friedel's salt, base exchange reactions (reactions between magnesium and portlandite, carbonation, alkali-silica reactions (reactions between alkali hydroxides and silica) and puzzolan reactions (reactions between calcium hydroxide and silica) were evaluated.

The need for an update of the concrete leaching process has been identified primarily as being of importance for the evolution of the water chemistry inside the concrete barriers as well as outside the barriers, see section 7.1.3. However such an analysis should of course also include an evaluation of the change in transport properties of the concrete barriers.

An update of the analyses of the other processes will probably be required only if the conditions of importance for the processes are different from those on which the former analyses are based. For example, if the new prognosis of the amounts of sulphate in the wastes is different from earlier prognoses an update of the consequences of formation of monosulphate/ettringite may be necessary. In addition, the result of the on-going projects aiming at improving the inventory of chemical substances in the wastes may show the need of new analyses of concrete alteration processes (Appendix A1).

Chemical alteration of gravel backfill and near-field rock

In the previous assessment no credit has been taken to the potential retardation in any gravel backfill or in the near-field rock. The influence of hydroxide leached from the concrete barriers on the physical and chemical properties of potential sand/gravel backfill and near-field rock should be studied in order to investigate the possibility of including the gravel backfill and near-field rock as barriers for radionuclides.

7.1.5 Interactions between repository parts

The dissolution and subsequent dispersion of barrier and waste components should be studied to investigate the potential for and effects of chemical interactions between different repository parts. For this, information on the magnitude and direction of water flow between the Silo and the caverns are needed, see section 7.1.1.

7.2 Far field

7.2.1 Geology

A geological structure model is mainly used as input to hydrogeological analyses. Axelsson and Hansen (1997) have reviewed the underlying structural model for the hydrogeological modelling carried out in the previous assessments. They compile evidence (fracturing on cores, seismic indications, and indication of hydraulic connections from interference tests) for the structures in the previous model.

They conclude that the previous model is plausible, but alternative interpretation of some of the major zones can be motivated. The alternative interpretation shown in Figure 7-1 as well as the previous one should be considered in subsequent hydrogeologic modelling and transport analyses.

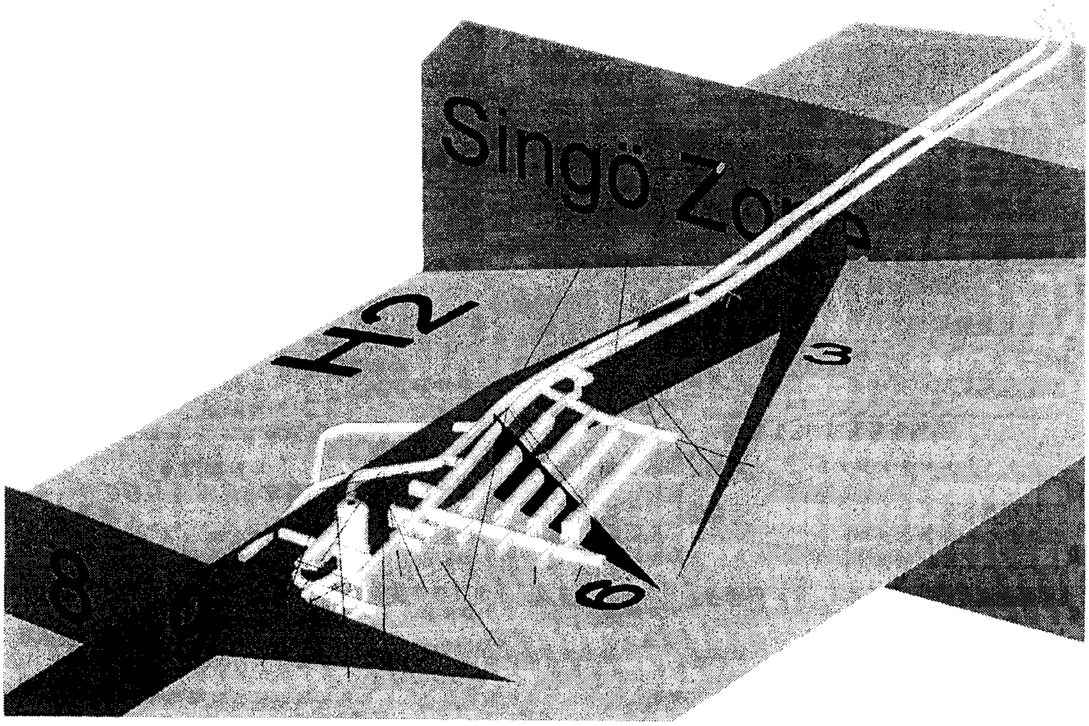

Figure 7-1: Updated geological structure model suggested by Axelsson and Hansen, (1997).

7.2.2 Hydrogeology

Objectives

A revised hydrogeological model of the SFR site should (Appendix A4):

- present a credible representation and understanding of the hydrogeological system,
- explore effects of seals and possible extensions of the repository,
- provide input to the quantitative safety assessment.

Evidently these main objectives are coupled. In order to meet these objectives the following strategy is suggested.

The modelling should be based on available information. Therefore an attempt will be made to calibrate the model on hydrogeological data measured before, during and after construction of SFR-1. Mainly data on inflows and time to breakthrough of Baltic water will be used since it is realised that many other data available are irrelevant or too uncertain to be considered.

Changes in boundary conditions due to land-rise have to be considered in the modelling. In addition, the consistency of the models and data has to be checked and the significance of uncertainty in models and data explored.

The closure of the SFR facility means that the tunnels need to be sealed in order to reduce the groundwater flow through the engineered system and to avoid chemical interactions between different vaults. Proper seals should be included in the modelling and it may also be necessary to explore where seals need to be placed in order to avoid too much flow. In analysing the seals they will be represented by a suitably low hydraulic conductivity. However, the evaluation on how to achieve such seals in practice is not a part of the modelling.

The results will be used as input both to the source-term calculations (both source term itself and judgement of risk of chemical interaction between vaults) and the geosphere migration modelling, but also to the biosphere assessment (temporal and spatial distribution of migration path).

Input to source-term (i.e. near-field transport) calculations

According to planned developments of the source term calculations (see sections 7.1.1 and 8.1) the following hydrogeological results (i.e. output from the hydrogeological modelling) would be needed:

- groundwater turnover and direction of flow in each individual rock vault (BLA, 1BTF, 2BTF and BMA) - where the hydraulic conductivity of the different vaults may vary in time due to degradation of the engineered barriers,
- separation of the total flow into components, i.e. distinguish between the flow in the rock surrounding the vaults and the flow in the different layers in the vaults (BMA and Silo) as principally displayed in Figure 7-2,
- estimation of flow mixing between different vaults, which may affect the judgement of chemical interactions between the vaults,
- the time for fresh water intrusion into the Silo and the vaults, which may affect the water composition in the near field.

It is also important to visualise the calculated groundwater flow on a vault scale to use as a reference for the analysis. As already noted the hydraulic properties of the vault internals (K_s, K_b etc. see Figure 7-2) may change with time due to degradation. Information on these changes would need to be obtained from the near-field evaluation within the SAFE-project, but the hydrogeological evaluation needs to be able to handle these changes.

Input to far-field migration calculations

As will be discussed in section 8.2 the following hydrogeological input would be needed for migration calculation:

- (distribution of) flow paths,
- Darcy velocity and flow wetted surface preferably integrated to "F-parameter"-values along each flow path (see e.g. Andersson et al., 1998),
- "non-sorbing" breakthrough time,
- dispersion estimates.

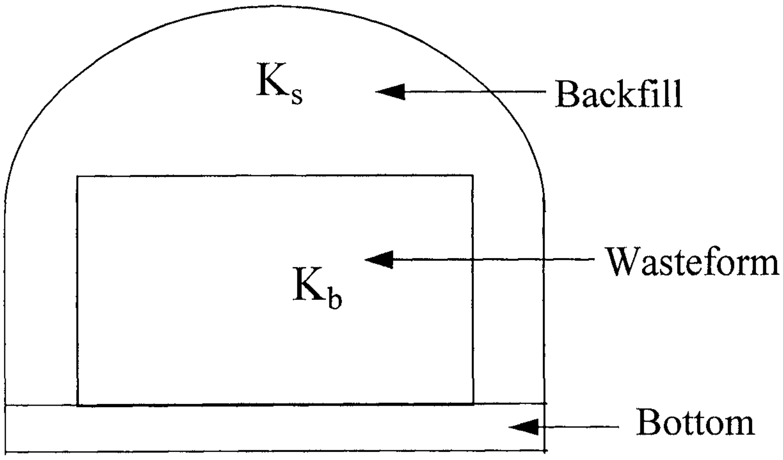

Figure 7-2. Schematic cross-section through the BMA. For the estimation of radionuclide release it is very important to determine whether the groundwater flows through the waste form, the backfill, the bottom part or through the backfill. Flow directions are also very important.

However, it should be kept in mind that the release flow paths at the SFR site can be quite short and that the rock mass is quite permeable. Uncertainties in the retention parameters are usually also quite large. This implies that the expected retention of the geosphere may be fairly limited for the SFR site. On the other hand the previous hydrogeological assessment of the inland scenario (Carlsson et al., 1987) resulted in very long release flow paths. If this is the case, far-field retardation may be significant, at least for some scenarios and in particular for the relatively short lived nuclides in the SFR inventory.

Input to biosphere assessment

The hydrogeological modelling should provide the following information to the biosphere modelling:

- an assessment of the spatial and temporal distribution of groundwater recharge/discharge,
- an assessment of the spatial and temporal distribution of migration release points (discharge areas),
- an assessment of the dilution in the geosphere (i.e. transverse dispersion) and well dilution (i.e. dilution of radionuclides in wells).

Impact from near-field and biosphere modelling

The hydraulic properties of the near-field barriers, engineered barriers and near-field rock, may vary in time due to different degradation processes. The near-field analyses and modelling has to provide information on such changes in hydraulic properties.

There are processes in the biosphere that also affect the hydrogeology, e.g. land rise and redistribution of sediment layers. Therefore, the biosphere modelling and the hydrogeological modelling should use consistent assumptions regarding topography, land rise and potential evolution (changes) of the top-sedimentary layer (as well as uncertainties in these properties).

Suggested modelling

It is suggested that the hydrogeological modelling is done on three scales. On the *regional scale* the focus should be on modelling the impact of important regional processes and features such as transient boundary conditions, variable density, major uncertainties in the structural interpretation. On the *repository scale* the focus should be to provide the input needed for the source term calculations within the safety assessment. The regional and repository scale models should also be used to identify the portion of rock where migration may take place, and to identify the regions for discharge into the biosphere. The *migration scale* should focus on resolving the groundwater flow field in order to produce detailed migration paths and migration properties of these paths.

Below, the suggested modelling is briefly described, focusing on the objectives. A more detailed description of the modelling is given in Appendix A4.

The main objective of the *regional scale analysis* is to find out which regional processes and features that governs the groundwater flow on more detailed scales and to provide boundary conditions for the repository scale analysis. Special attention will be given to:

- evaluating to what extent (if at all) varying density effects need to be considered,
- evaluating the significance of uncertainties in the regional structure model,
- identification of a suitable domain for the repository scale and migration scale models,
- determination of changing boundary conditions for the repository scale model (and possibly for the migration scale model) as well as describing the evolution of groundwater recharge and discharge areas under these changes,
- providing input to the evaluation of the evolution of the geosphere groundwater chemistry.

Apart from the direct delivery of boundaries and boundary conditions, the regional model may potentially be used as a basis for motivating simplifications in the process description of the repository scale model (such as omitting density effects and/or division into time periods with more stable small-scale flow). Initial analyses with two-dimensional models in order to explore significance of density effects has already been done and the results indicate that the density effects are small compared to the effects of land rise (Stigsson et al., 1998).

The objectives of the *repository scale analysis* is to evaluate the groundwater flow situation on the repository scale under varying conditions and thereby to:

- provide estimates of the groundwater flow needed in the source term modelling,
- identify the portion of rock where far-field migration may take place, and to identify the regions for discharge into the biosphere (provided this region falls within the repository scale model),
- provide estimates of the evolution of groundwater discharge and recharge areas in the repository scale,
- evaluate the effect of wells and the possibilities for dilution in wells placed in the repository region,
- evaluate the effect of plugs on the flow in the vaults,
- assess the uncertainties in the predictions, which are caused by the uncertainties in the adopted modelling approach.

In meeting these objectives it would be necessary to represent all tunnels and rock vaults directly in the model while keeping the representation of the main aspects of the heterogeneity of the rock.

The objectives of a *migration scale analysis* are to resolve the spatial variability of the groundwater flow such that it captures essential migration properties. Such an analysis would provide:

- estimates of the migration flow paths and the hydrogeologically related migration parameters needed for calculating radionuclide migration in these paths (see section 8.2),
- estimates of the distribution of biosphere release points (see section 8.3).

The location and size of the migration scale model will depend on the outcome of the regional and repository scale models. Three different approaches for detailed groundwater flow modelling are explored in the "alternative model project" (Ström and Selroos, 1997): a stochastic continuum, a discrete fracture network and a channel network approach. These approaches should be considered when selecting a model concept for deriving the hydrogeologic input to the migration modelling in SAFE.

7.2.3 Geochemistry

The future chemical composition of the far-field groundwater may in principle affect the sorption characteristics to apply when modelling far-field radionuclide migration (see section 8.2). Consequently, it is concluded in Appendix A4 that careful geochemical evaluation of the site would only be necessary if more credit is placed on migration in the geosphere. The groundwater chemistry would then possibly impact the selection of sorption characteristics (see section 8.2). To a large extent the issue of future groundwater chemistry will be connected to the outcome of the regional groundwater flow analysis. Consequently, if it is concluded that further importance should be given to migration modelling, there will be a need to also evaluate the groundwater chemistry, but such an evaluation should be co-ordinated with the regional groundwater modelling.

The groundwater composition may affect the chemical composition of the water during an initial period after repository closure. The required information on groundwater composition should be able to get from the SFR control program on water chemistry without any additional geochemical evaluation (see section 7.1.3). In a long-term perspective the saline-fresh water interface may reach the repository vaults and affect the chemical conditions in the near field. It is therefore of interest to estimate the time required for fresh water breakthrough in the near field.

7.2.4 Gas

The issue of gas production and gas escape from the repository should be reconsidered in a scenario and process analysis of SFR. However, previous studies (Thunvik and Braester, 1986) strongly suggest that the gas transport capacity of the rock strongly will exceed the potential gas production in the repository. It appears that gas migration in the rock will still remain a minor issue.

7.2.5 Rock mechanics and temperature

At present, there are no indications that there is a need to analyse rock mechanical effects and temperature effects. However, the development of a system description model (see Chapter 6) may alter this view.

7.3 Biosphere

7.3.1 Introduction – Objectives

In previous safety assessments the biosphere had a minor role or it was described in a stylised way. However, the coming regulations may require that the effects should be described for a realistic biosphere. In addition, potential effects on other organisms than man should be evaluated. This requires a more realistic model of the biosphere based on ecological know-how rather than isolated simplified food-pathways. Presented models must be clearly related to the surrounding natural system to receive common acceptance.

The concept with concentration factors currently used in the biosphere modelling is questionable due to high variations introduced. A more appropriate way is to use trophic transfer models or at least subdivide the concentration factors in a physical-chemical and a biological component.

To be able to construct a trophic transfer model of the biosphere in the SFR analyses a better knowledge of the biological structure of the system is necessary. Suggested actions to achieve this are:

- Use more site-specific information. The area is well studied since the construction of the nuclear power plants.

- A better description of geomorphology and bottom sediments should help in selecting adequate processes and parameter values.
- A careful study of the area in combination with hydro-geological modelling should help to identify the entrance points to the biosphere of contaminated ground water.
- Local accumulation zones should be studied.

7.3.2 Structure and flow of loose deposits

Sigurdsson (1987) performed a detailed geological field survey of the sea floor above SFR. This detailed information about sediment depth and particle size gives important information on residence time, sorption and transport pathways, but has not been used in further estimates. Moreover this detailed information is useful to estimate particle transport due to resuspension. The topography and hydraulic conductivity of the material is also an important input to the hydrogeological model.

There is a need to consider the sediment distribution in a regional scale (i.e. Öregrundsgrepen). There is also a need to evaluate the impact of erosion and resuspension by waves, ice and wind in particular in combination with land rise. The resuspension may affect accumulation of radionuclides in sediments and also cause rapid releases of accumulated activity. It may also affect topography, which in turn may affect the groundwater flow in the rock (see section 7.2.3).

7.3.3 Water turnover

The water turnover was identified as an important factor for the dose-estimates. The water turnover also affects the transport and dispersion of radionuclides in solution and bound to particles and organisms. A new method to calculate the water turnover will be used in the area (Engqvist, 1997). The turnover will also be recalculated at different stages during the land rise. The effects of migrating organism and various foodstuffs will be incorporated in the radionuclide transport calculations, see section 8.3.

7.3.4 The structure and succession of the ecosystem

The structure of the ecosystems surrounding SFR, with emphasis on the aquatic systems needs to be described. Probably the most important communities will be filter feeders, deposit feeders and macrophytes and fish. In addition, the total carbon content in sediment and water is important to estimate in order to be able to describe the carbon flow and the dilution effects of ^{14}C, see section 8.3. Information is available from previous studies and from the monitoring programs around the Forsmark nuclear power station, but additional field surveys may also be necessary.

After the ecosystem structure of today is described, the succession to future states must be predicted. Land uplift is the most dramatic change that is possible to predict. Some extreme variations need to be tested in order to account for uncertainties in other future changes that are difficult to predict, e.g. changes in climate and salinity of the sea, acidification of lakes and land use.

7.3.5 Interaction geosphere/biosphere

At the interface between the geosphere and biosphere important processes occur that have not been addressed in earlier safety assessments. Moreover, the near surface hydrogeology needs further consideration since it affects the transport of radionuclides in the loose deposits. In the time-scales and physical distance regarded here, there will be a considerable feedback from the biosphere to the geosphere, e.g. trophic status affecting water chemistry. These issues can only be resolved with an integrated approach between the geosphere and biosphere.

7.3.6 Eco-system model

An eco-system model, which describes the transfer of radionuclides in the biosphere, will be developed using the basic information described above. This model is further discussed in section 8.3.

8 Radionuclide Transport

Identified improvements concerning the prediction of radionuclide transport in the near field, far field and biosphere and resulting dose to man and the environment are summarised in this chapter. The issues addressed concerns both modelling approaches and improvements of models as well as selection of data. More details are given in Appendices A5 and A6 and to some extent in Appendix A3.

Also here the work with the development of a system description model may display additional needs not yet identified, see Chapter 6 and Appendix A2. It may also show that some of the suggestions given in the following sections are of less importance for the release and migration of radionuclides in the SFR waste.

8.1 Near field

8.1.1 General

This section is focussed on suggestions regarding improvements of the modelling of the near-field radionuclide migration in terms of models and underlying assumptions and input data. In evaluating the need of improvements in models and input data there are some aspects that should be considered. These aspects are related to the relationship between the input parameters and the modelling results and the relationship between the time constants of the migration processes in the system and the half-lives of the radionuclides. These issues are further described in Appendix A5.

8.1.2 Assumptions and data

In the previous assessments a "conservative" approach was used in the modelling of radionuclide release and migration in the near field, in terms of both the assumptions made and the data selected in the quantitative analyses. If a "best estimate" approach is to be added in the forthcoming assessment, more information to support a realistic choice of assumptions and selection of data is needed.

Radionuclide dissolution and migration properties

In the former assessments it was assumed that all radionuclides were immediately dissolved and evenly distributed within the interior of the different repository parts at the time of repository closure. In addition, the potential sorption capacity in waste and barrier materials in BLA was neglected. To be able to make more realistic assumptions and selection of data the following suggestions are made.

- The time required for water saturation of the repository barriers after repository closure affects the time when radionuclide dissolution starts. A time delay could be of importance for short-lived radionuclides, especially in the Silo with its bentonite barriers and for waste conditioned in bitumen.

- Making waste type specific considerations instead of averaging the radionuclide content in each repository part may allow for more detailed source term analyses, such as accounting for radionuclide solubilities and limited release from waste conditioned in bitumen. However, this requires data on radionuclide inventory, waste composition, conditioning materials and waste allocation on a waste type level.
- The possibility of taking credit for the available sorption capacity in the waste and barrier materials in BLA could be investigated. The effect of filling the tunnel with for example sand (if possible from a mechanical point view) may also be worthwhile to study.
- Data on radionuclide solubility, sorption, and diffusivity in concrete materials, bentonite and rock materials should be updated according to the most recent information available. However, the consistency in data compared to similar SKB assessments must also be considered.
- A more realistic choice of potential changes in barrier properties and near-field conditions with time than in earlier radionuclide release modelling. In the previous assessment two distinct time periods were considered, the salt-water period and the inland period, depending on the location of the shoreline. In the modelling it was assumed that the barrier properties and near-field conditions changed momentarily at the start of the inland period and the data were conservatively chosen in order to be representative for the whole inland period.

Release of radioactive gas

Release of radionuclides as gas can be a fast transport mechanism to the biosphere. Therefore, it is suggested that potential mechanisms for obtaining radionuclides in gaseous form in the repository should be investigated, e.g. formation of gaseous ^{14}C by microbial degradation of organic materials.

8.1.3 Models

In the previous assessment, a simplified description of the internal parts of the Silo and the caverns was used in the modelling of radionuclide release from the near field. For example, the transport resistance in waste matrices was not considered, but the sorption capacity of conditioning cement was taken into account. The increased amount of bitumen as conditioning material for the waste implies that the transport resistance in the waste matrix could be of importance. It is therefore suggested that a more detailed description of the internal parts of the Silo and the caverns (except BLA) are included in the models, at least for time periods when no major changes in transport properties are expected (see Appendices A3 and A5). One alternative is to use available models for water uptake and subsequent release of radionuclides from bitumen conditioned waste for separate analyses of the effects on the source term release (see Appendix A3).

The need for a more detailed modelling at longer times when the barriers are more degraded could be investigated by the use of simple models (see Appendix A5).

8.2 Far field

8.2.1 Objectives and factors to consider

The modelling of the transport through the geosphere will essentially depend upon the distribution of migration paths in rock and the migration properties along migration paths (see also Appendix A5).

Migration paths

The migration paths will be determined, modelled, in the migration scale hydrogeological analyses (see section 7.2.3). The regional scale and repository scale analyses will be used to judge whether migration paths will change with time as a consequence of land rise. It is possible that different distinct periods can be identified during which the migration paths would be stable, such as the marine period and the inland period considered in previous analyses, but it is not certain and provisions for handling such potential complications in the migration modelling may be needed.

Migration properties along flow paths

The transport in the geosphere of radionuclides that diffuse into the rock matrix and are sorbed in the matrix is mainly determined by the water flow-rate distribution and the flow wetted surface area. For nuclides that are not sorbed in the matrix, the flow porosity may also influence the transport depending on the residence time in the geosphere. The retardation of radionuclides transported in a fracture zone may be small because of the large water flow rates. On the other hand, the retardation could be large due to the possibly large flow-wetted surface areas found in a fracture zone. Even in short paths with a small flow rate and a large flow wetted surface area, retardation by matrix diffusion and sorption may be important.

As discussed by Elert (1997) and further elaborated by Andersson et al., (1998) there is a general lack of data for the flow wetted surface and its potential correlation to the important migration paths. This implies that rather generic data, with wide uncertainty ranges, would need to be applied.

There is also a need for matrix porosity, diffusivity and sorption data. However, precise values are not very critical. It will probably be possible to apply the values suggested for SR 97.

Migration models

The main migration modelling approach used in the SKB deep repository analyses such as SR 97 is to divide the flow field into multiple single flow paths and to adjust the one-dimensional advection dispersion matrix diffusion model FARF31 (Norman and Kjellbert 1990) to each migration path. The migration scale hydrogeological analyses (see section 7.2.3) will probably apply the stochastic continuum concept, the discrete fracture network concept, or the channel network concept. A combination of approaches may also be considered. The potential benefits of these different approaches are currently evaluated within the framework of SR 97 and it is recommended that the approach within SAFE should build on experiences reached within SR 97. Andersson et al., (1998) show how to derive input from any of these different detailed groundwater flow analyses to equivalent parameters to be used in FARF31.

An alternative to migration calculation in FARF31 could be direct solution of the migration in the migration scale hydrogeology codes. Such possibilities already exists within the discrete fracture network model (code: Fracman-PAWork) (Miller, 1996) and the channel network model (code: CHAN3D) (Gylling, 1997).

The methodology to be used to model the radionuclide transport in the geosphere may be chosen based on the results of the SR97 project.

8.3 Biosphere

8.3.1 Introduction

The potential radiological impact of the near-field/far-field releases is evaluated in the biosphere radiological analysis. This analysis includes both a modelling of the radionuclide migration in the biosphere and calculations of the radiological impact in terms of dose. The analysis needs input primarily from the far-field migration calculations.

For the analysis of the migration within the biosphere an improved ecosystem model is suggested (see Appendix A6). This model builds on the description of the biosphere already discussed in Section 7.3 and in Appendix A6. In addition, the following suggestions are made:

- the turnover of ^{14}C in aquatic systems should be evaluated in more detail,
- the exposure to flora and fauna from released radionuclides should be considered,
- model data and dose-conversion factors should be reviewed and updated,
- all assessments should be given with estimates of the uncertainties in the results.

8.3.2 Ecosystem model

The transfer of radionuclides in the biosphere will be described in an ecosystem model. The aquatic part of the ecosystem model is shown in Figure 8-1. It is subdivided in a physical and a biological sub-model. The physical model (Figure 8-1, upper) describes sedimentation, resuspension and water-exchange mainly affected by physical processes. The radionuclides will enter the system through the sediment. In the sediment interface there is a drastic change in redox conditions, bacterial activity, temperature etc. which affects the retention of radionuclides.

The ecological part of the model (Figure 8-1, lower) is more complex. The model should attempt at describing the food (and thus radionuclide) transfer between the different components of the ecosystem such as suspended particles (plankton and detritus particles), filter feeders (e.g. mussels), fish and birds. However, several of the compartments (the boxes) can be omitted or aggregated when the structure of the system is described (see Section 7.3 and Appendix A6). There are already models (Gilek et al, 1997; Kautsky, 1995) developed for assessing the turnover of hydrocarbons (DDT, PCB, Dioxin) through filter feeders. These models can easily be adopted to calculate the turnover of radionuclides.

The benefits with ecosystem models are that the transfer of radionuclides is dependent on the food transfer and not on concentration coefficients. This means that the large range of variation of the concentration coefficient is reduced. In addition, transient changes in the environment can be modelled. The ecosystem model is also more transparent for other scientists because it describes existent processes and pathways.

A similar model is necessary for the terrestrial environment. However, this model has fewer compartments and is thus easier to construct.

8.3.3 Effects on non human biota

The coming regulations from SSI stresses that effects on other organism than man should be estimated. From the ecosystem model described above, compartments will be identified which have high process-rates and storage of radionuclides. The radiation effects in these compartments will be calculated. Moreover, two top-carnivores will be selected to illustrate the potential effects of bio-magnification. Two well known and suitable top-carnivores are grey-seal and white-tailed eagle. These are occurring in the area and are well investigated concerning organic xenobiotic substances (e.g. PCB, DDT).

8.3.4 Evaluation of dose consequences

The ecosystem model will predict where high levels of radionuclides will accumulate. This enables that radiation effects to organisms living in the area can be calculated. This will also give new ideas of potential pathways not considered earlier. This information will be incorporated in the development of the BIOPATH model to calculate doses to man. The results from the data collection and field surveys will give the major structure of the ecosystem and estimates of future changes. The models of sediment resuspension and landscape change give information on transport of particles and input data to the far-field model. This affects the dispersion and transport of radionuclides together with the water turnover.

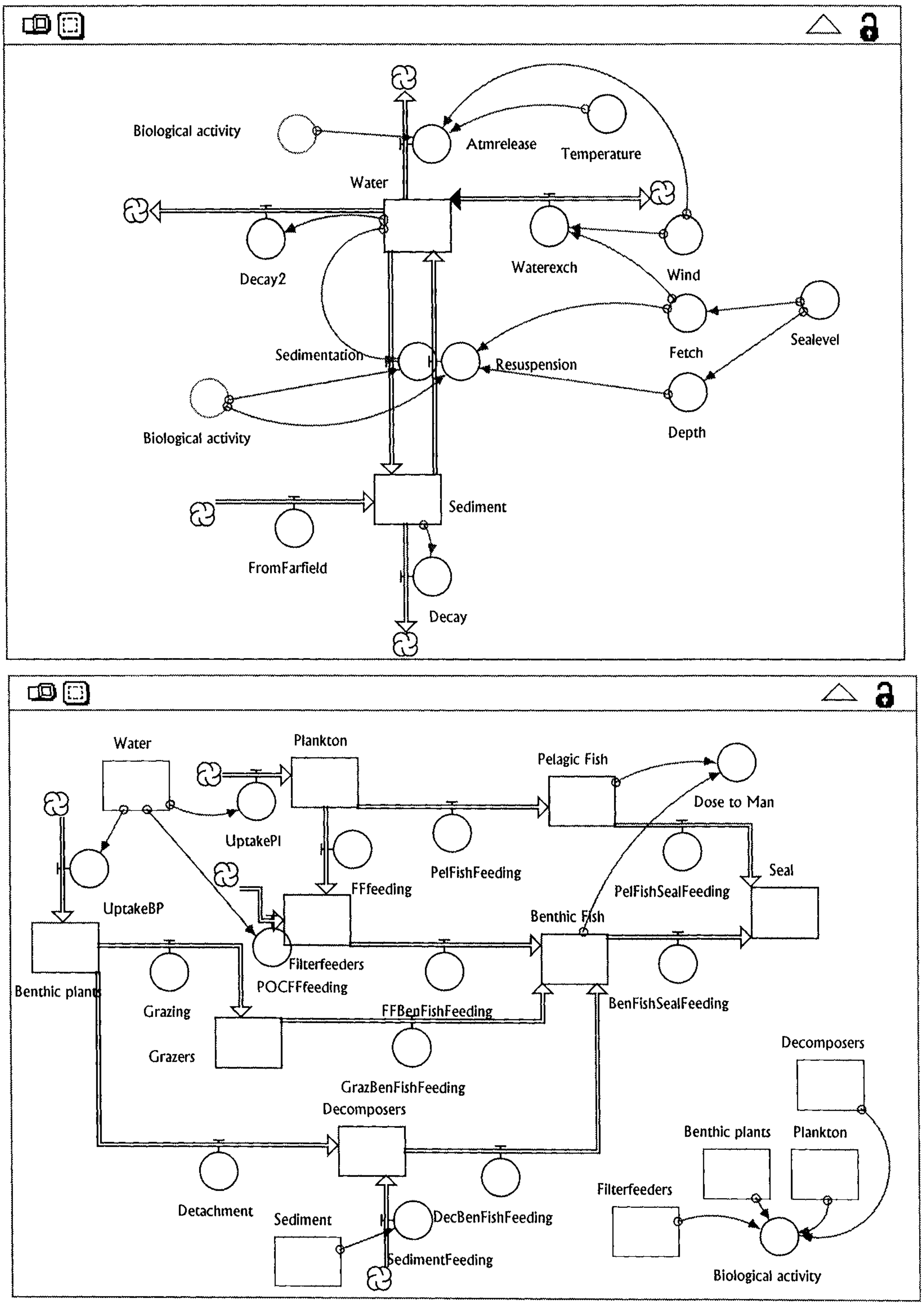

Figure 8-1. Conceptual ecosystem model of an aquatic ecosystem in terms of physical processes (upper) and biological processes (lower). Boxes are reservoirs, thick arrows are fluxes of matters, and thin arrows are relationships to parameters and external variables (circles).

9 Conclusions and Recommendations

The observations and recommendations compiled in this report are the result of a re-evaluation of the previous safety assessments and the supporting reports, taking the comments and remarks given by the authorities in their review of the previous safety reports into account. These results are summarised in this chapter.

The recommendations given below will primarily be used as input in developing a detailed work plan for phase 2 of the SAFE project. The aim of phase 2 is to provide models and data to continue to the third and last phase of the project. The aim of the third phase is to perform release calculations and to write the Final Safety Report.

9.1 General Recommendations

An important question is how to handle model selection and identification of information flow. An information flow diagram is useful in order to identify the interaction between models, data etc.

A related issue is how to address the question of transients in the system. In the previous assessment this question was simplified by analysing two distinct stages of the long-term evolution, one marine and one inland case. In the SAFE project this approach will be abandoned and a more transient approach will be used.

Quality control is important in the SAFE project. The project leader is responsible for the "official data" and the transfer of data between different parts of the project. A data "freeze" should be made before the start of the third phase of the project.

The data that should be used in the calculations, and the choice of models, should be compiled in a report. A similar report (Wiborgh et al 1987) was published in the safety assessment that took place in 1988.

9.2 Inventory

The main interest in the inventory part is to assemble basic data on the wastes. Recommended reports to be published are:

- A prognosis of waste volumes, waste materials and amounts of radionuclides. This can be co-ordinated with the ordinary prognosis work for SFR-1.
- A review of the radionuclide inventory focused on how to calculate the "hard-to-measure nuclides" i.e. correlation factors or other ways to calculate radionuclides. The review should be based on existing reports.

In order to write these reports there is a need to collect data on materials and chemicals in the SFR waste.

9.3 Scenarios

In the previous assessment the approach with a reversed event tree was used for systematic scenario analysis. Since then, the methodologies for scenario construction have developed and improved. The suggestion for the SAFE-project is to use the interaction matrix method. The approach consists of a series of steps.

- Construction of interaction matrices including auditing against international FEP-lists.
- Prioritisation of the processes and interactions in the matrices.
- Identification of scenario initiating FEPs and scenarios.

The interaction matrix should be used to check whether processes and interactions judged to be important actually are considered in the selection of calculation cases and models for the quantitative analyses. In addition, it should be used to judge the relevance of potentially important scenario initiating events and conditions.

The proposed work with the development of a system description model in the form of interaction matrices may well display additional needs of analyses not identified here. Therefore this work should be carried out early in the project in order to be able to incorporate the result of the work in an updated project plan at an early stage.

9.4 Near Field

The reports regarding the near field should address several issues.

- There is a need to collect data and describe the present state of knowledge of degradation processes that could occur in SFR-1. Examples of such processes are degradation of concrete and bentonite, corrosion and swelling/dissolution/ageing of bitumen. The rates of the processes are of great interest.
- Gas generation and gas transport are important processes in the repository. The work suggested is to find or develop a model as a foundation to calculations, but it is noted that there is already much done in this field. Gas transport in the far field should also be addressed in this work.
- A review of the reactions between radionuclides and complexing agents that could occur in the repository should be made. This work should include an update of the state of knowledge regarding the degradation of cellulose and how this will affect the radionuclide solubility and the sorption of radionuclides in the barriers in SFR.
- The extension of a zone with high ionic strength and high pH of the water outside the concrete barriers, the so-called pH-plume, and how this could affect the properties of the rock and backfill should be investigated.
- A review of microbiologic processes that could affect the barrier performance should be made.
- All numerical modelling can be performed in much greater detail now than in 1988. A repository scale hydrogeological model should be developed, see section 9.5. The model should be detailed enough to be able to describe flow within and between

individual tunnels. Different plugging of the repository should also be analysed with the hydrogeological models.

9.5 Far Field

The suggested work in the far field part of the project is to develop models for geology and hydrogeology. The recommended hydrogeological modelling are divided into three scales, regional scale, repository scale and migration scale. The recommended modelling work is summarised below.

- A review of the interpretation of geological structures should be performed in the project (already done).
- The regional scale model should find out which regional processes and features that give effect on the groundwater flow in the repository scale model. It should also provide the repository model with boundary conditions. Special attention should be given to effects of varying density, to the identification of suitable domains for the repository and migration scale modelling and to the determination of changing boundary conditions.
- The repository scale model is described in the near-field recommendations, see section 9.4.
- If the project should develop a migration scale model depends on the result of the repository scale model. If the main flow paths are too short there is no need to perform the migration scale modelling. If the length of the flow paths are judged to be sufficient enough to have an effect on the radionuclide release, the modelling should give estimates on the detailed migration flow paths and parameters needed for the calculations as well as on the distribution of biosphere release points.

A 3D-CAD model has been developed. This greatly simplifies the suggested three-dimensional hydrogeological modelling.

9.6 Radionuclide Migration in Near Field and Far Field

Much of the work in the second phase of the SAFE project is aimed to provide data or models to perform the release calculations in phase 3. In phase two, release calculation models should be modified to fit the SFR calculation cases and test calculations should be made. In addition, the model for gas release through the Silo barriers should be improved during phase 2.

9.7 Biosphere

The work in the biosphere field includes a larger degree of development than the other fields. The reason is that in previous safety assessments the biosphere was described in a stylised way. The coming regulations require that the radiological effects should be described for a realistic biosphere, including also potential effects on other organisms than man.

The following work is recommended.

- An ecosystem model for the area should be developed. The model should be based on site-specific information. The area is well studied since the construction of the Forsmark nuclear power plants, which should be compiled.
- A better description of morphology and bottom sediments should help in selecting adequate processes and parameter values. It should also evaluate effects of sediment redistribution.
- Model studies of local accumulation zones should be performed.
- A detailed evaluation of the turnover of ^{14}C in the aquatic systems should be performed.
- Dose-conversion factors should be updated.
- The exposure to non-human biota from released radionuclides should be considered.

9.8 Operation and Incidents

An update of the analysis of possible incidents should be made.

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