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Project SAFE

Update of the SFR-1 safety assessment Phase 1

Appendices

October 1998

Svensk Kärnbränslehantering AB
Swedish Nuclear Fuel
and Waste Management Co
Box 5864
SE-102 40 Stockholm Sweden
Tel 08-459 84 00
+46 8 459 84 00
Fax 08-661 57 19
+46 8 661 57 19

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Foreword

This report contains the Appendices to the report on the first phase of the SAFE project, SKB Report R-98-43. The appendices are:

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Appendix A1:

Inventory

Per Riggare

SKB

Contents

1 INTRODUCTION

In order to calculate the long-term performance of a waste repository one has to start with a waste inventory. Therefore one can say that the waste inventory is the basis for all safety assessment.

In the licenses (SSI 1988 (b), SKI 1988 (b)) for the SFR-1 facility it is said that a renewed safety assessment should be carried out at regular intervals. In SSI 1988 (b) the interval is specified as at least each ten years. The latest safety assessment was published in 1993 (SKB 1993) which was an update of the 1987 final safety report (SKB 1987). In order to fulfil the demand in the license SKB has started a project; SAFE (Safety Assessment of Final Repository for Radioactive Operational Waste) which will carry out a renewed assessment which will be finished in year 2000.

The aim of this study is to identify the possible improvements in the waste inventory compared to former assessments. In the reviewed literature special attention has been directed towards the authorities review of the assessments.

In the SAFE project the waste inventory is defined as the amount of waste, waste matrix, engineered barriers and other construction materials which will be left in the repository at the time of closure.

In this report the inventory is divided in four parts, "general", "waste", "waste matrix" and "construction materials".

2 GENERAL

It is always possible to make waste inventory, or to use another word a source term, by making some rough estimate. The hard part is to make the uncertainties as small as possible. The aim for the inventory part of the SAFE project should be, with a reasonable cost, to make the uncertainties as small as possible.

2.1 REALISTIC VS. CONSERVATIVE INVENTORY

In reviewing the waste inventory of SFR-1 one notices there are significant differences between the inventory that has been used in the radionuclide transport calculations and what the prognosis over radionuclides, volumes and so on show. The reason is of course that in the designing of SFR-1 there was some safety margin included.

Recent work show that the margins are between prognosis and the design inventory is increasing. The reason are mainly the improvements that the nuclear power plant and other waste producers has been doing in their waste treatment and the continued low frequency of fuel failures. The use of shallow land disposal for VLLW has also meant much for the decreasing volumes.

The ambition of the SAFE project is to use two principle different cases for the radionuclide transport calculations. A conservative (pessimistic) case to use as official limiting case and a realistic case which should be used to calculate the probable future development of SFR-1. This means that two inventories must be defined.

The conservative inventory should (if possible) not differ from the one in SKB 1993 and SKB 1987. Changes can be done if the need for raised limits occur in some case or if the calculations show that some limit must be decreased in order to make the safety case. Both developments are unlikely to happen.

The realistic inventory should be based on the best knowledge at present time. Much of the work has already been done in different prognosis and annual reports to the authorities.

The approach with conservative and realistic inventories is applicable for mainly for radionuclides. For chemicals and other materials it is harder to state what a conservative inventory is.

The existing inventory has a level of detail with different inventories in the different rock vaults. In some cases this is a conservative assumption, in other cases a non-conservative assumption. Today, with the better computer technology and the new computerised database of the SFR-waste, there is a good opportunity to make a better differentiation, so the inventory can be divided on a waste type level. The ambition in the SAFE-project is to make radionuclide transport calculations where the starting point is the "waste type level".

In the end it is possible that one is not capable or does not have the need to do so detailed calculations, but the aim of the inventory should be to provide data detailed enough to make the detailed calculations possible.

2.2 WASTE VOLUMES

The volumes of waste are not interesting in it self for the long-term performance of SFR-1 as long as there are available space in the repository. The amounts or volumes of different waste types are nevertheless interesting as the foundation for calculating the amounts of radionuclides, chemicals and other materials.

The volumes of waste that is stated in SKB 1993 are mainly from a prognosis made in 1987. If one compares with the latest prognosis (Riggare 1995) one can conclude that there is room for great improvements regarding the relative amounts of different waste types

Since 1987 several changes has been made:

- Forsmark NPP nowadays uses only bitumen as matrix for stabilisation of ion-exchange resins.

- Ringhals NPP now uses a shallow land disposal at the site for very low-level waste.

- Barsebäck NPP has stopped sending their burnable low-level waste to Studsvik for incineration.

- New waste types, e.g. ashes from pyrolysis and cement solidified sludge, has been approved for disposal.

- No consideration of large scrap components was taken in SKB 1993.

In short, the inventory of waste amounts can be improved and if the SAFE project shall carry out realistic calculations it is a definite condition in order to calculate the amounts of nuclides, chemicals and other materials.

The last prognosis was made in 1995 and the next planned is to be done in 1998. In order to get the best possible estimate of future production the work should be performed as late as possible. Since the safety report should be published in 2000, it seems that the next prognosis should be postponed from 1998 to 1999. Until the new prognosis is finished the 1995 one should be sufficient for all foreseen needs in both project SAFE an in the ordinary work with SFR-1.

3 WASTE

The waste part of the inventory is quite complex. In order to give a more stringent review, this chapter is divided in three parts, "Radionuclides", "Chemicals" and other "Other materials".

3.1 RADIONUCLIDES IN SFR-1

The total allowed activity amount is 10^{16} Bq according to the licenses (SKI 1987 (b), SSI 1987 (b)). There are also limits on different nuclides, see Table 3-1.

Table 3-1. Allowed nuclide inventory in SFR-1.

Nuclide	Half-life (yr.)	Silo (Bq)	BTF (Bq)	BMA (Bq)	BLA (Bq)
^{3}H	12.3	$1.3 \cdot 10^{14}$	-	-	-
^{14}C	$5.7 \cdot 10^{3}$	$6.8 \cdot 10^{12}$	$1.3 \cdot 10^{11}$	$2.9 \cdot 10^{11}$	$2.6 \cdot 10^{9}$
^{55}Fe	2.7	$7.1 \cdot 10^{14}$	$1.7 \cdot 10^{13}$	$1.0 \cdot 10^{14}$	$2.3 \cdot 10^{12}$
^{59}Ni	$7.5 \cdot 10^{4}$	$6.8 \cdot 10^{12}$	$1.5 \cdot 10^{11}$	$1.0 \cdot 10^{12}$	$2.3 \cdot 10^{10}$
^{60}Co	5.2	$1.8 \cdot 10^{15}$	$4.0 \cdot 10^{13}$	$2.6 \cdot 10^{14}$	$5.8 \cdot 10^{12}$
^{63}Ni	100	$6.3 \cdot 10^{14}$	$1.5 \cdot 10^{13}$	$8.8 \cdot 10^{13}$	$1.9 \cdot 10^{12}$
^{90}Sr	28.8	$2.5 \cdot 10^{14}$	$2.7 \cdot 10^{12}$	$6.5 \cdot 10^{12}$	$7.1 \cdot 10^{10}$
^{94}Nb	$2.0 \cdot 10^{4}$	$6.8 \cdot 10^{9}$	$1.5 \cdot 10^{8}$	$1.0 \cdot 10^{9}$	$2.3 \cdot 10^{7}$
^{99}Tc	$2.1 \cdot 10^{5}$	$3.3 \cdot 10^{11}$	$3.6 \cdot 10^{9}$	$8.8 \cdot 10^{9}$	$1.1 \cdot 10^{8}$
^{106}Ru	1	$6.1 \cdot 10^{12}$	$6.2 \cdot 10^{10}$	$1.7 \cdot 10^{11}$	$2.1 \cdot 10^{9}$
^{129}I	$1.6 \cdot 10^{7}$	$1.9 \cdot 10^{9}$	$2.2 \cdot 10^{7}$	$4.7 \cdot 10^{7}$	$6.4 \cdot 10^{5}$
^{134}Cs	2.3	$8.1 \cdot 10^{14}$	$1.1 \cdot 10^{13}$	$2.2 \cdot 10^{12}$	$2.6 \cdot 10^{11}$
^{135}Cs	$3.0 \cdot 10^{6}$	$1.9 \cdot 10^{10}$	$2.2 \cdot 10^{8}$	$5.3 \cdot 10^{8}$	$6.4 \cdot 10^{6}$
^{137}Cs	30.2	$4.9 \cdot 10^{15}$	$5.3 \cdot 10^{13}$	$1.3 \cdot 10^{14}$	$1.4 \cdot 10^{12}$
^{238}Pu	87.7	$1.2 \cdot 10^{12}$	$1.7 \cdot 10^{10}$	$3.1 \cdot 10^{10}$	$4.7 \cdot 10^{8}$
^{239}Pu	$2.4 \cdot 10^{4}$	$3.8 \cdot 10^{11}$	$6.9 \cdot 10^{9}$	$1.2 \cdot 10^{10}$	$1.9 \cdot 10^{8}$
^{240}Pu	$6.6 \cdot 10^{3}$	$7.8 \cdot 10^{11}$	$1.1 \cdot 10^{10}$	$1.9 \cdot 10^{10}$	$2.9 \cdot 10^{8}$
^{241}Pu	14.4	$4.2 \cdot 10^{13}$	$5.4 \cdot 10^{11}$	$9.4 \cdot 10^{11}$	$1.5 \cdot 10^{10}$
^{241}Am	433	$1.0 \cdot 10^{12}$	$1.3 \cdot 10^{10}$	$2.4 \cdot 10^{10}$	$3.8 \cdot 10^{8}$
^{244}Cm	18.1	$1.2 \cdot 10^{11}$	$1.5 \cdot 10^{9}$	$2.8 \cdot 10^{9}$	$4.4 \cdot 10^{8}$
Total:		$9.3 \cdot 10^{15}$	$1.4 \cdot 10^{14}$	$5.9 \cdot 10^{14}$	$1.2 \cdot 10^{13}$

The latest prognosis over radionuclides content in SFR-1 was made in 1995 (Riggare 1995). The prognosis show that it is most probable that the 10^{16} Bq limit will probably never be achieved in SFR.

The existing inventory origins in the design (Thegerström 1981) and has not been changed for either SKB 1987 or SKB 1993. The reason is of course that it is a conservative inventory with considerable safety margins. There are considerable margins, according to the latest prognosis (Riggare 1995). The margins for both ^{60}Co and ^{137}Cs are factor a five. For the different plutonium nuclides the final margin is expected to be between a few up to thirty per cent. These margins are a well-known fact, this fact is mentioned in SKB 1987, SKI 1988, SSI 1988, SKI 1992, SSI 1992 and SKB 1993.

Nevertheless one should remember that the radionuclide prognosis is based on the assumption that all Swedish nuclear power plants are in use until 2010 and that the fuel failures are on a continuing low level.

As mentioned in Section 2.1 there should be two inventories. The one described above is the conservative one and should not be changed unless there is a need for it. The realistic one on the other hand must be based on some sort of prognosis of the radionuclides. It seems a good idea to do this at the same time as the prognosis of volumes in 1999.

All following suggested improvements refer to the realistic radionuclide inventory.

3.1.1 General comments

All nuclides are not equally easy to measure. Hard emitting gamma emitters like ^{60}Co and ^{137}Cs are very easy to measure, which means that it is very easy to keep track of the amounts in SFR-1. On the other hand, alpha and beta emitting nuclides can not be measured on the final waste package. Since these nuclides, especially the alpha emitters, are of great interest for the long-term performance of the repository, one has to make some indirect estimate.

One way to make this estimate is to use correlation coefficients or correlation factors, i.e. the relation between an easy to measure nuclide and the hard to measure nuclide. The coefficient can be generic or specific, for example power plant specific. Correlation factors are used for several nuclides in SFR-1. In SKB 1987 and 1993 there are general coefficients for ^{55}Fe, ^{63}Ni, ^{14}C, ^{94}Nb, ^{59}Ni, ^{106}Ru, ^{134}Cs, ^{90}Sr, ^{99}Tc, ^{135}Cs and ^{129}I towards ^{60}Co and ^{137}Cs. For transuranic nuclides (TRU) and ^{90}Sr plant specific factors are used. The generic factors are calculated in Thegerström 1981.

All correlations have a quite large uncertainty and new improved measurements and changed plant suggest that a review of the factors should be performed. Unpublished material from the SKB project "Other waste" also shows that a review should give better correlations.

3.1.2 Specific nuclides

^{60}Co and ^{137}Cs

The nuclides ^{60}Co and ^{137}Cs are mentioned in SSI 1988, SSI 1992 and SKI 1992. These two nuclides will dominate the radionuclide inventory in the initial state and will in certain scenarios give a contribution to individual doses to the critical group. No additional investigations of the nuclides are needed at present since they are very easy to measure with gamma spectroscopy on the waste package. These nuclides are also measured by routine.

^{14}C

^{14}C is a nuclide that has been commented in SKI 1988, SSI 1988, SKI 1992 and SSI 1992. ^{14}C is a pure beta-emitter, and estimation of the activity from measurements on the waste package is therefore not possible. It is also hard to estimate ^{14}C since the correlation factor is very poor (Thierfeldt 1995). Another feature is that it is very hard to estimate the fraction of produced ^{14}C from the nuclear power plants that comes with the waste to SFR-1.

The chemical speciation is of great interest since organic and inorganic ^{14}C have very different physical and chemical properties. For example the sorption properties on concrete differ between the organic and the inorganic species.

The reviews mentioned above also point out that the ^{14}C-inventory is very uncertain but they also states that the safety margin is wide enough. At the same time the reviewers also says that SKB should try to acquire better knowledge in the subject.

The conclusion is that the ^{14}C should be reviewed, especially the questions of the fraction of ^{14}C that comes to SFR-1 and the chemical speciation.

^{59}Ni and ^{63}Ni

Nickel-59 and nickel-63 are two hard-to-measure nuclides that are mentioned in SKI 1992 and SSI 1992. Especially ^{59}Ni is interesting since it has a half-life of 75 000 years and thereby makes it significant in the long run dose-perspective.

A research program is currently working. The aim is to acquire a more sensitive measuring method by using Accelerator Mass Spectroscopy (AMS).

^{59}Ni is the limiting nuclide for metal components close to reactor core. This makes the nuclide interesting from an economical point of view. It is more expensive to deposit these scrap components in SFL than in SFR-1. It is therefore of great interest to take as much as possible of these waste components to SFR-1. A good estimate of the ^{59}Ni in the ion-exchange resins is then needed in order to find out how much of the limiting (conservative) inventory that must be set aside for ^{59}Ni in resins and to make better measurements of the near-core scrap components.

^{36}Cl

Chlorine-36 is a nuclide that has not been a nuclide of interest in the former safety assessments. ^{36}Cl is very hard to measure since it is a pure beta emitter. SKI 1988 and SSI 1988 highlights this and states that the inventory is between 1-100 GBq, based on the chlorine content in the reactor water. SSI 1988 says that the expected inventory of ^{36}Cl is quite low and should not give any significant contribution to the collective or individual doses.

Even if the expected doses from ^{36}Cl are small SKB should try to make a good estimate.

TRU and ^{90}Sr

Some other hard-to-measure nuclides are the TRU-nuclides and ^{90}Sr that are not directly measurable on the waste package. ^{90}Sr and the TRU-nuclides have great radiotoxicity.

Since these nuclides give a substantial part of the long-term release to the environment they are of great interest. A special program has been working since 1988 with the aim to increase the knowledge and to have better control of the TRU and ^{90}Sr inventory. There is nowadays a large database over TRU-nuclides and ^{90}Sr which can be used for more detailed assessment. For example there is a need to see how these nuclides are distributed between the different rock vaults in SFR and how to calculate the relations between nuclides that are hard or impossible to separate in the analysis, e.g. $^{239}Pu/^{240}Pu$ and $^{241}Am/^{238}Pu$. This work can be performed within the existing programme.

3.2 CHEMICAL SUBSTANCES

In this report "Chemical substances" are defined as chemically pure substances as, 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.

Other substances, for example different metals, concrete, plastic and wood is defined as materials.

In general the waste in SFR-1 is well characterised and there are only small amounts of chemicals.

Some references to the safety reports (SKB 1987, SKB 1993) discuss questions about chemical substances. The main focus has been towards complexing agents, potentially environmental hazardous substances and some other chemicals that are well known to be in the waste like sulphate and borate/boric acid.

The increasing environmental awareness in the last years has influenced the whole society towards new, less environmentally hazardous, chemicals. This change of chemicals and the use of them have also affected the nuclear power plants. For example the power plants have exchanged their old detergents to new ones. These detergents often contain strong complexing agents, which if they are deposited in SFR-1 can seriously disturb the sorption properties of some radionuclides. Since these detergents and possibly some other substances can find their way into the waste there is a need to get better knowledge and SKB has therefore started a project to identify and, if the need arises, quantify chemicals in the waste.

3.2.1 Complexing agents

Strong organic complexing agents like EDTA, citric acid and oxalic acid can seriously disturb the sorption properties for many radionuclides and should therefore be avoided in SFR. (Complexing agents formed by degradation of cellulose are addressed in Section 3.6). All waste producers are aware of this fact and only minor amounts are deposited in SFR-1.

SKB should try to get better knowledge in this field since the complexing agents are used in decontamination and present in some detergents. The use of decontamination will probably increase in the future. Specifically there is a need to have a better knowledge about the amount of complexing agents that can be deposited in SFR-1 and what the consequences will be.

3.2.2 Potential environmental hazardous chemical substances

According to Scandiaconsult 1982 and Meijer 1987 the environmental impact is very small from environmentally hazardous substances. No action is foreseen except the above-mentioned investigation of chemicals in SFR waste.

3.2.3 Other Chemicals

There are chemicals in the waste that have no hazardous properties, but are still interesting due to some other property. For example enhanced deterioration of concrete by sulphate.

Concrete additives

Concrete additives are a group of chemicals of which increased knowledge is required, both regarding what kind of chemicals they contain and the effects they may have on repository performance. SKB has, on demand from SKI, started a project that aims to increase the knowledge in this field.

Borate / boric acid

Borate is discussed in SKB 1987 and SKB 1993. Borate / boric acid is used in PWR and ends up in certain waste types. Borate can already in small amounts affect the hardening of the concrete. This is of slight interest since it is neutralised with lime before solidification.

No action is foreseen in this area.

Sulphate

Sulphate is a species of interest since it can damage the concrete structure. An inventory of the amounts of sulphate was made in 1987 (Wiborgh 1987). The prognosis has changed since 1987 and therefore a simple check should be made to assure that the amounts are accurate.

3.3 OTHER MATERIALS

Two processes that can enhance the transport of radionuclides are complexation of nuclides and production of gas.

The inventory of complexing agents is discussed in 5.3.1 except for the degradation products of organic substances. The dominating contributor for complexing agents is cellulose. In an alkaline environment, cellulose will degrade under release of iso-saccarinic acid (ISA). ISA is a strong complexing agent with the strongest complexation to metal ions of the valence of three or four, e.g. different plutonium ions.

Gas is produced from three sources: corrosion of metals, microbial degradation of organic matter and radiolysis of water. The dominating source is the corrosion.

There are target values in SFR-1 on materials that can produce complexing agents and gas production, see Table 3-2 and 3-3. The gas production can, with some assumptions, be transformed into metal areas subject to corrosion, see Table 3-4. The assumptions are corrosion rates of 1 mm/year for aluminium and zinc and 3 μm/year for steel. These assumptions should be checked against the literature. The target values can be exceeded if an approval by SKI is given.

The inventory in SFR-1 is based on information in the waste type description on the average composition. Especially the packages with scrap metals and refuse is important in these estimates. Recent reports (Riggare 1997) show that these average compositions are inaccurate and a new improved average composition is presented. The uncertainties are still large in this area and a further look into this question should be beneficial.

Table 3-2. Limits of different materials in SFR-1.

Material	Silo (ton)	BMA (ton)	BTF (ton)	BLA (ton)
Ion-exchange resins	1500	1100	1600	50
Bitumen	920	1300	-	150
Cellulose	20	64	20	1000
Sludge	-	50	40	-
Other organic material	-	125	-	1000

Table 3-3. Limits on gas production in SFR-1.

Process	Silo (Nm^3/yr.)	BMA (Nm^3/yr.)	BTF (Nm^3/yr.)	BLA (Nm^3/yr.)
Corrosion	1700	4000	1500	3000
Radiolysis	50	-	-	-
Microbiologic activity	5	-	-	2000

Table 3-4. Calculated limits on gas producing materials.

Material	Silo	BMA	BTF	BLA
Steel (m^2)	$1.2 \cdot 10^5$	$3.1 \cdot 10^5$	$1.2 \cdot 10^5$	$2.3 \cdot 10^5$
Aluminium / Zinc (m^2)	450	1200	450	900

4 WASTE PACKAGE AND MATRIX

The waste packages and the matrix are well defined. In SKB 1993, SKB 1987 and in the waste type descriptions there are thorough descriptions of the waste package and types of matrix. With an accurate prognosis over different wastes there is no problem to estimate the amounts of materials the package and waste contribute with.

5 CONSTRUCTION MATERIALS

Only the construction materials that are left at time of repository closure is of interest in the safety assessment. The dominating materials that are of any concern will be the concrete constructions in the rock vaults, the shotcrete in the vaults (and possibly in the tunnels) and the backfill in the vaults and in the tunnels. The concrete additives may be of importance (see Section 3.5.3).

In Wiborgh et al 1987 there is a thorough account for the amounts of construction and barrier materials. This reference shows both a realistic inventory and also account for the inventory used for calculations. What is not in the reference is the amount of reinforcement bars. The near field assessment will show if there is a need to make an estimate of the content of iron in the reinforcement.

6 SUMMARY / CONCLUSIONS

One of the aims in the safety assessment of SFR-1 is to make radionuclide calculations to estimate the release to the environment. In order to make these calculations there is a need to be able to describe the inventory in greater detail. The improvements in computer technology and the new computerised database of waste in SFR-1 gives a good possibility to achieve this.

The aim for project SAFE is to make both conservative (pessimistic) and realistic radionuclide transport calculations. To achieve this goal there must be two inventories, a conservative and a realistic inventory. The conservative inventory is the inventory used in the design of the repository, which in most parts are identical with the limits in the licence for SFR-1. This inventory should not be changed unless there is a very strong need for it. Since it is not likely that there will be any changes in the conservative inventory, all changes discussed below regards the realistic inventory.

Volumes

Although the waste volumes are not interesting in themselves, the volumes are the foundation of the inventory. Therefore there is a great interest to have good estimates of the volumes of the different waste types. A thorough prognosis should be made in 1999, but until then the latest one from 1995 could be used in calculations.

Radionuclides

The total (actual) inventory of nuclides is calculated from the measurements of the easy-to-measure nuclides since, in principle, all hard-to-measure nuclides are calculated by correlation factors to ^{60}Co and ^{137}Cs. These factors should be reviewed since there are quite large uncertainties involved.

There are also some specific nuclides where better knowledge should be an advantage.

- ^{14}C is a nuclide that dominates the individual doses after a few hundred years and the collective dose in the inland-scenario. The amount of the nuclide is uncertain since the correlation factor is very uncertain. The chemical speciation of ^{14}C is also of interest due to different properties of organic and inorganic carbon.

- ^{36}Cl is a nuclide that is very hard to measure. Although the authorities in their reviews of the safety reports say that there probably are small doses from chlorine, the inventory should be improved.

- ^{59}Ni is a long-lived nuclide that limits the close-to-the-core metal scrap that can be taken to SFR-1. There is an ongoing research project that aims to provide a better measuring method. This should make it possible to improve the knowledge about ^{59}Ni inventory. The assumption that 90 % of the inventory is collected in the ion-exchange resins should be checked.

- TRU / ^{90}Sr are subjects to a special program. The database from this work should be used to calculate a better (more detailed) inventory. Calculations of the amount of different plutonium nuclides in SFR-1 should be done and an estimate of how TRU-nuclides are distributed in the repository vaults should also be performed.

Chemicals

Chemical substances, both in the waste, the waste package and in the construction materials are of great interest since even small amounts can seriously disturb the long term performance of SFR-1. SKB has already started a project to make a better inventory of what chemicals there are in SFR-1. There is also a special program about concrete admixtures going on. These two projects will hopefully improve the knowledge of the chemical substance inventory of SFR-1.

Other materials

Since the construction of SFR-1 and the materials used are well known there are no needs for improvement. The waste package 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 volumes is.

Regarding knowledge of the inventory of the actual waste there is no need for action for ion-exchange resins waste. On the other hand there is room for improvement in the refuse and scrap metal inventory. A better average composition of different materials should decrease the uncertainties.

A literature review of corrosion rates of steel, aluminium and zinc in an alkaline environment should be performed.

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Project SAFE – Prestudy

Appendix A2:

Scenarios

Kristina Skagius
Marie Wiborgh

Kemakta

Contents

1 INTRODUCTION

1.1 BACKGROUND AND AIM

SFR-1 is a final repository for low and intermediate level radioactive waste from reactor operation. The repository is located in rock beneath the Baltic Sea at Forsmark and has been in operation since 1988 when the application for operation was approved by the authorities. The licence for operation of SFR-1 contained requirements on some complementary investigations concerning the performance of the silo, of which one was to carry out and present a consistent and logic scenario analysis.

To meet this request from the authorities, SKB initiated a study with the aim to identify and formulate scenarios for the long-term performance of the silo as well as the vaults in SFR-1 (Skagius and Höglund, 1991). The methodology used in this study was influenced by the methodology outlined in a joint SKI/SKB scenario development project (Andersson et al., 1989). The main steps in this methodology are:

- the identification of Features, Events and Processes (FEPs) that may influence the performance of the repository,
- a screening of the identified FEPs as being part of the Process System (PS) or acting outside the PS,
- a structuring of the FEPs belonging to the PS to obtain a logical description of the PS, and
- formulation of scenarios by applying FEPs acting outside the PS onto the PS.

The Process System, PS, was defined as "the organised assembly of all phenomena (FEPs) required for description of barrier performance and radionuclide behaviour in a repository and its environment, and that can be predicted with at least some degree of determinism from a given set of external conditions".

The scenario formulation study and results of the quantitative analyses of the scenarios were summarised in a safety report in 1991 (SKB SFR 91-10). Since then, the methodologies for systematic scenario construction have developed and improved.

SKB are now in the position of updating the safety analysis of SFR-1 and have for that purpose set up project SAFE. The project is divided into three phases. The first phase is called a prestudy, and the aim 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. The second phase is devoted to the accomplishment of the studies proposed in Phase 1, and a new safety analysis will be carried through during the third phase of the project.

The project is also divided into different topics of which Scenarios is one. The main objective for the topic Scenarios in Phase 1 of the project is to propose a systematic methodology for the selection and description of scenarios to be considered in the safety analysis of SFR-1. A secondary objective is to update the list of FEPs that was compiled as a part of the scenario identification and formulation study carried out in 1991.

1.2 OUTLINE OF THE APPENDIX

In order to meet the objectives of Phase 1 of project SAFE, topic Scenarios, the methodology used for scenario identification and formulation in the previous safety assessment of SFR-1 and other methodologies used for similar purpose are reviewed. The methods are described in Chapters 2 and 3 in this Appendix. Based on experiences from applying these methodologies, a methodology for identifying and constructing scenarios to be used in the new safety assessment of SFR-1 is proposed in Chapter 4.

The terminology used in this appendix is given in a glossary in Attachment A to this appendix. These definitions are also proposed for the future scenario work in project SAFE. An updated version of the FEPs list is given in Attachment B.

2 METHODOLOGY USED IN PREVIOUS SAFETY ANALYSIS OF SFR-1

2.1 GENERAL

The methodology used in the previous safety analysis of SFR-1 for identification and formulation of scenarios was influenced by the methodology outlined in a joint SKI/SKB scenario development project (Andersson et al., 1989). The purpose of the study was to identify and formulate scenarios for the long-term performance of the repository in terms of release and transport of radionuclides from the disposed waste to the biosphere. The methodology contained the following steps:

- a compilation of an initial FEPs list containing phenomena which potentially could influence the long term performance of the repository
- a selection of FEPs which were judged to belong to the Process System from the initial FEPs list
- the development of a logical description of the Process System by graphically displaying how FEPs within the Process System are linked according to cause and effect and in text describe the FEPs and links between FEPs
- a selection of scenarios and formulation of these scenarios as the time evolution of the Process System.

These steps are briefly described in the following sections and specific parts where improvements would be beneficial are pointed out.

2.2 IDENTIFICATION OF FEPs AND SELECTION OF FEPs BELONGING TO THE PROCESS SYSTEM

The initial FEPs list was obtained by going through the FEPs compiled in the joint SKI/SKB scenario development project (Andersson et al., 1989). FEPs which were assessed to be specific for a spent fuel repository according to the SKB concept were excluded, e.g. FEPs related to the performance of a copper canister and of no relevance for the type of waste and barriers in SFR-1. Additional FEPs were identified and added to the list by people involved in the safety assessment of SFR-1.

No formal documentation of the FEPs was done explicitly for this application. FEPs selected from the SKI/SKB scenario development project were described in memo-text in the published report of this project, and new FEPs were named in such a way that it should be clear from the name which phenomena it represented. An attempt was made to prepare a formal protocol for each FEP on the list. This protocol included:

- a general description of the FEP,
- the cause and effect of the FEP,
- a screening of the FEP as belonging to the Process System or acting outside the Process System or being screened out,
- other FEPs that this FEP could be grouped or connected with, and
- references to the literature.

Filling in these protocols was found to be very time consuming, and because of this and because of the lack of a documentation system easy to work with the protocols were not completed for all FEPs on the list. This meant that the selection of the FEPs to be included in the Process System was done during the construction of the graphical description of the Process System by using the initial FEPs list as a checklist.

This lack of documentation of the FEPs is one weakness of the methodology as applied in the previous safety analysis. Today computerised databases are used for this purpose and a number of FEP databases has been created, both in Sweden and in other countries. It should therefore be possible to improve both the identification of relevant FEPs and documentation of FEPs by utilising available FEP databases and creating a computerised SFR database.

2.3 DESCRIPTION OF THE PROCESS SYSTEM

The Process System was described both in graphical form and in text explaining the visualisation. The visualisation of the Process System was made in a reversed event-tree structure, where the main branch consists of the release of radionuclides from the initial source through the different barriers and the top event is the release of radionuclides from the geosphere to the biosphere (Figure 1). The diagram was created by starting with the top event and then moving inwards barrier by barrier towards the initial source, the waste matrix, linking FEPs together in branches according to cause and effects. A branch ends either with a phenomena influenced by external conditions, i.e. FEPs acting outside the Process System, or with basic information. Basic information was defined as given prerequisites of the system (initial conditions), such as repository design and waste composition.

In constructing the diagrams it was found that links between events and processes in general had to go via properties of the physical components of the system. These types of FEPs were then introduced even if they were not found in the initial FEPs list. Examples are "Physical properties" and "Chemical properties" of different barrier materials and "Water composition" in different parts of the repository. It was also found that it was not possible to describe the Process System in a reversed event-tree structure without considering the FEPs representing water composition in the repository as being basic information, i.e. FEPs in the bottom of a branch that are not affected by other FEPs. This is of course not the case since the water composition in the repository is affected by chemical and transport processes in the repository, and this simplification that had to be introduced is then a draw-back of the visualisation method.

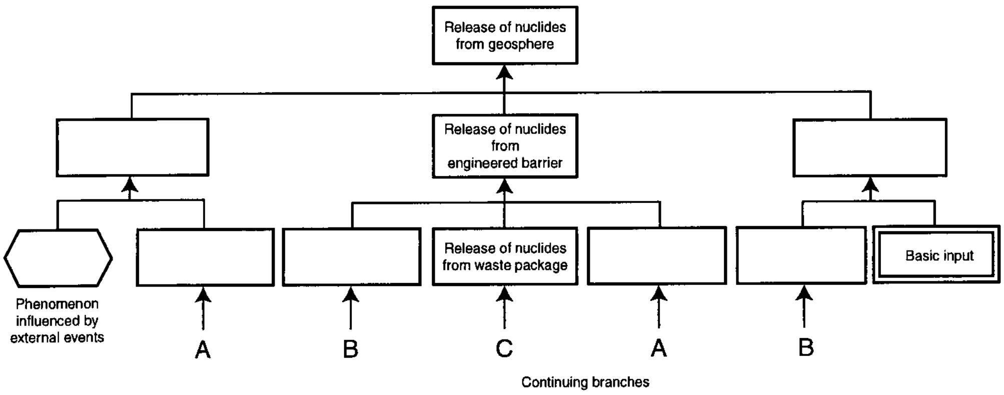

Figure 1. Structure of the reversed tree of events

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 the text. In this screening process, the graphical description was an important aid for checking the logic and consistency of the screening.

Applying the reversed event-tree structure in the safety analysis of the SFR-1 was the first attempt within the SKB in the development of a method for obtaining a structured and visualised description of the Process System. Since then other methods have been developed and tested (see Chapter 4). In retrospect it could be concluded that the reversed event-tree structure is very well suited for visualisation of events or chains of events with a well-defined beginning and end. However, it is less advantageous as a visualisation tool for systems where the evolution of the system to a great extent is caused by ongoing processes.

2.4 SELECTION AND FORMULATION OF SCENARIOS

The scenario methodology applied in the safety analysis of the SFR-1 contained no formal procedure for selection of scenarios, i.e. for selection of FEPs not belonging to the Process System, but when imposed to the Process System forms a scenario for the evolution of the process System. The main reason for this was that the objective of the scenario work was to check and verify the description of scenarios already selected and analysed in the safety assessment preceding the licensing in 1988 (SSR, 1987), rather than to identify new 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. The scenario initiating FEPs for the extreme scenarios were fracturing of the concrete silo and blocking of the gas release paths in the silo and a combination of these, and these were applied on the Process System for the reference scenario. The extreme scenarios for the vaults concerned the drilling of wells directly into the vaults.

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 the different scenarios. This was carried out by addressing processes directly affecting radionuclide migration, and also how these processes are affected by the physical and chemical properties of the barriers and their evolution in time.

2.5 REGULATORY REVIEW

The deepened safety assessment of the SFR was reviewed by the authorities SKI and SSI (SKI and SSI, 1992). These authorities assessed the scenario work to be satisfactorily carried out in that the most important scenarios were identified and described. However, some criticisms were given to the coupling between the scenario work and the selection of calculation cases.

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 SKI and the SSI was 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 making variations in models and parameter values.

3 OTHER METHODOLOGIES APPLIED FOR SCENARIO CONSTRUCTION

3.1 GENERAL

The methodology based on a reversed tree of events for structuring of FEPs in the Process System was the first approach tested by SKB to systemise and visualise the Process System. Since then two additional methodologies for scenario construction have been tested by SKB, namely Process Influence Diagrams (PID) and Interaction matrices. These two methodologies are briefly described below.

3.2 PROCESS INFLUENCE DIAGRAMS (PID)

3.2.1 Introduction

A methodology based on Influence Diagrams with linked documentation was developed by SKI in the SITE-94 project (Chapman et al., 1994; Chapman et al., 1995; SKI, 1996). This methodology was tested as a scenario construction tool in the prestudy of the repository for long-lived, low and intermediate level waste, SFL 3-5 (Skagius and Wiborgh, 1994 and Wiborgh (ed.), 1995). The methodology as applied to the SFL 3-5 repository concept involved the following main steps:

- Construction of a basic or general version of the PID
- Development of a scenario specific PID from the basic or general version
- Formulation of Scenarios and calculation cases.

3.2.2 Construction of a basic or general version of the PID

The Basic Influence Diagram should ideally contain all FEPs that are relevant for the system studied for any scenario. This requires a definition of the system in terms of waste form, barrier design and materials as well as repository layout. In addition, decisions must be taken on the geometrical extension of the system to be included in the PID, i.e. the extension of the Process System. The aim of the assessment will have to determine where the system boundary is set and to what level of detail the repository components should be described in the PID.

To facilitate construction of the PID, the Process System can be divided into regions, e.g. representing different barriers in the disposal concept. After identification and compilation of all FEPs belonging to the Process System they are introduced into the PID as boxes with a FEP name in each of the regions where they could occur. Within each region, interactions between FEPs are identified and represented on the PID by arrows linking pairs of FEPs and showing the direction of the influence. There are no restrictions in the number of interactions between two FEPs since one FEP may influence another FEP in several, different ways. Each influence arrow is designated with a unique code.

A more comprehensive description of the FEPs and definition of the interactions are given in documents stored as separate records in a database. Each record is electronically linked to the corresponding FEP-box or influence-arrow in the PID.

3.2.3 Development of a scenario specific PID from the basic or general version

The Process System in the basic or general version of the PID contains FEPs and influences that may affect the behaviour of the repository system, but at this stage no evaluation are made of the importance of these on the repository performance. Since the significance of influences and FEPs may depend on the initial conditions of the Process System and on how the Process System is affected by the surroundings, these entities must be defined; i.e. the scenario premises or scenario initiating FEPs must be selected.

The significance of each influence for the selected scenario premises or scenario initiating FEPs are then assessed by "expert judgement", using a pre-defined scale of significance. The assessed significance of an influence is together with explanations of the decisions documented in a protocol that is electronically linked to the arrow representing this influence in the PID. The result of the significance assessment can also be displayed in the PID by the use of colour coding or different line types and fillings for different significance levels.

Reduced scenario specific PIDs can now be prepared at different significance levels by removing influences assessed to be of lower significance than the defined level. A FEP can only be removed if all its influences on other FEPs or from other FEPs are below the defined significance level.

3.2.4 Formulation of Scenarios and calculation cases.

The reduced scenario specific PIDs are used to formulate the scenario and to identify calculation cases needed to analyse the scenario. Also at this stage, the documentation is essential as information source both for forthcoming studies and for review purposes. In order to make the documentation easily accessible the protocols linked to the PID can be used for recording how influences and FEPs are considered in the models and assessment calculations.

3.3 INTERACTION MATRICES

3.3.1 Introduction

In the Interaction Matrix methodology, the basic device in the Rock Engineering Systems (RES) approach, the interaction matrix, is used as a tool for identification and structuring FEPs in the Process System. The RES methodology was originally developed for approaching rock engineering problems (Hudson, 1992) but was tested by SKB as a potential method for structuring of FEPs relevant for the disposal of radioactive waste (Stephansson and Hudson, 1993; Stephansson and Hudson, 1994).

A comparison of the PID and the RES methodologies revealed that the methodologies in general are very similar (Eng et al., 1994). The main difference concerns the means of structuring and visualising the Process System. The matrix structure is less complex which facilitates the construction of the Process System and makes the Process System easier to present. However, the resolution of a matrix is generally lower than of a PID. This can to some extent be taken care of by increasing the number of diagonal elements in the matrix or by using several matrices to represent the Process System. The overall conclusion from the comparison of the methodologies was that a combination of parts of the PID and RES methodologies was likely to be a promising approach.

As a result of this comparison of methodologies a combination of these methods was tested by SKB as a part of the preparation for the forthcoming performance and safety analyses (Skagius et al., 1995). In this method the basic devise in the RES approach, the interaction matrix is used for identification and structuring of FEPs in the Process System, and the procedures for assigning priorities to interactions and for documentation are adopted from the PID methodology. These different parts of the methodology and some further developments that were made based on the experiences from the performed work are briefly described below.

3.3.2 Construction of the interaction matrix and assigning priorities to the interactions

The basic principle of the interaction matrix is to list the main features or properties of the system along the leading diagonal elements of a square matrix. The interactions between these main features or properties defined in the diagonal elements occur in the off-diagonal elements. This is illustrated in Figure 2 together with the clockwise convention for the influence direction.

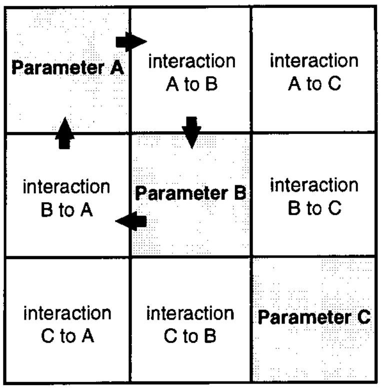

Figure 2. Principle of the interaction matrix.

Before starting the construction of the matrix, the objective of the assessment and the system to be covered by the PS must be defined, since these definitions have implications on the selection of diagonal elements in the matrix. The system definition includes a specification of the physical components of the repository system to be included in the Process System, the spatial extension of the Process System and the initial and boundary conditions of the system. Once this is done, the diagonal elements are selected and the features introduced in each diagonal element are defined and documented. To be able to describe relations between the Process System and the system outside the Process System, the boundaries of the Process System can be part of the leading diagonal elements of the matrix.

If the system to be studied and the corresponding matrix are large it may be practical to divide the matrix into sub-matrices. In such cases the overlap between the sub-matrices as well as the way the sub-matrices communicate with each other should be clearly defined.

When the leading diagonal elements in the matrix are specified and documented, the interactions between these main features are identified and described by introducing FEPs into the appropriate off-diagonal elements (interaction boxes) in the matrix. All interactions should be binary i.e. they should be direct interactions between features in two diagonal elements and not a path via a feature in a third diagonal element. Each interaction is documented by defining the process or event involved in the interaction as well as the features or properties in the two interacting diagonal elements that are influencing and affected by the process or event. Every off-diagonal element in the matrix should be checked for interactions, and the reason for not having identified any interactions in empty off-diagonal elements should be documented.

The next step is to set priorities to all identified interactions in the interaction matrix by the use of "expert judgement". The importance of the interactions is judged for the previously defined initial and boundary conditions, using a well-defined and documented priority scale. In order to facilitate later review and re-evaluations, a motivation to the assigned priority is given and documented together with the competence of the person or group of persons making the judgement. A colour coding can be used to display the priorities in the interaction matrix.

Both the identification of interactions and the setting of priorities may reveal requirements on modifications of the definitions of the diagonal elements in the matrix. Building the interaction matrix is therefore an iterative process.

The structuring of the Process System and the ranking of interactions require input from various information sources covering a broad range of disciplines. Therefore, these actions are preferable done by a group of people with both a general overview of the system and expertise in specific areas.

3.3.3 Documentation system

During the work with the testing of the interaction matrix methodology a documentation system in database format was developed. The database program FileMaker PRO was used for this purpose, and the documentation system contains two types of databases, one for FEP descriptions and one for matrix specific information. The reason for separating matrix specific information from more extensive descriptions of FEPs is that the same FEP may be involved in different interactions and the same FEP may occur in different parts of the repository system. The matrix documentation will then focus on the actual aspect of a FEP that is involved in a specific interaction, while the FEP documentation will contain a more general description of the FEPs. This so called SKB FEP database can then be utilised by several projects.

The *interaction matrix database* contains the matrix with names of the diagonal elements, representative for the features or properties of the diagonal elements, and colour-coded off-diagonal elements showing the highest priority of an interaction in an element. The database also contains all the different documents defining the:

- Objective of the assessment
- Studied system
- Different leading diagonal elements in the matrix
- Interactions between diagonal elements and assigned priorities with motivations.

The matrix and the different types of documents listed above are accessible via a menu system. In addition, the data records containing the description of the diagonal elements and the interactions between diagonal elements are linked to the matrix. This means that these records can be reached directly from the matrix, which facilitates the search of specific information. It is, of course, also possible to search for specific information directly in the document records without going via the matrix.

In the *SKB FEP database* general FEP descriptions with reference to the literature and to other assessments or FEP-lists are compiled.

The interaction matrix database and the SKB FEP database are coupled via cross-references. Each document record describing an interaction in the interaction matrix contains a reference to the appropriate FEP description in the general SKB FEP database. Likewise, each document record in the SKB FEP database contains a reference to the interaction matrix and the off-diagonal element where this FEP is found.

3.3.4 Formulation of scenarios and calculation cases

The interaction matrix with its linked documentation can be used to formulate the scenario or scenarios for which it was developed and to define calculation cases to be analysed. Its main function in this context is to serve as a checklist to ensure that all aspects assessed as important are addressed in the analysis of a scenario. The interaction matrix database can also be used to store information concerning the treatment of the interactions in the quantitative analyses of the scenarios.

4 METHODOLOGY PROPOSED FOR SCENARIO CONSTRUCTION IN PROJECT SAFE

4.1 INTRODUCTION

One experience from the testing of the three methodologies described in this Appendix is that the application of a new method requires development of procedures and tools, which in turn makes the work resource intensive. Therefore, only the three methods already tested by SKB are considered in this proposal of methodology for project SAFE.

The PID and the interaction matrix methodologies have been shown to be more suitable than the method based on a reversed tree of events as a tool for structuring the Process System. The main reason for this is that the evolution of the properties in a repository and its environment mostly is due to continuous, long-term processes that in turn are dependent on the evolution of the properties. This type of system is possible to describe both by influence diagrams and by interaction matrices, but not so well by a reversed tree of events, which are more suitable for describing the consequences of a series of events.

As has been shown above, the PID and the interaction matrix methods are quite similar. However, the interaction matrix method has been more extensively tested by SKB, which has resulted in the development of procedures and a documentation system that can be used in future applications. To be able to use already existing procedures and documentation systems will most likely reduce the resource need, and this would then be in favour for the interaction matrix method.

Another advantage with the interaction matrix method is that the whole procedure of constructing the matrix and assigning priorities to the interactions can be done together by a group of people. This increases the possibility of finding all interactions in the system. This is also the case for constructing a PID, but the actual construction of the PID involving the identification of interdependencies between FEPs would be more difficult to do in a group.

Based on the arguments above it is proposed that the interaction matrix method be used for the scenario work in project SAFE. It will allow the whole project group to participate in the work, and this will facilitate the information transfer in the project and, hopefully, also give all project members an increased understanding of the behaviour of the whole repository system.

The proposed procedure is described below.

4.2 DESCRIPTION OF PROPOSED METHODOLOGY

4.2.1 Scenario selection

It is proposed that the scenario work is divided into the following two parts:

- Development of an interaction matrix description of the Process System for a Reference Scenario
- Identification of the effects of alternative scenarios on the behaviour of the Process System by imposing Scenario initiating FEPs (EFEPs) onto the interaction matrix description of the Process System for the Reference Scenario

This means that the initial and boundary conditions for a Reference Scenario should be defined as well as Scenario initiating FEPs (EFEPs) for selected alternative scenarios.

The interaction matrix methodology involves, so far, no formal procedure for scenario selection in order to ensure that all relevant EFEPs are identified. This may not be needed if scenarios could be seen as a means of illustrating *possible* future behaviour of a system rather than *predicted* future behaviour. However, the reasons for choosing certain scenarios should be documented.

It is proposed that the project group do the final selection of scenarios. Going through existing FEP-lists may aid in this process. As a first input to the decision on scenarios, it is here suggested that the Reference Scenario is based on realistic initial and boundary conditions, i.e. consider expected initial states of the barriers and include land rise. Based on the opinion of the regulators after reviewing the previous safety analysis of SFR it is also suggested that drilling of wells near the repository should be considered either in the Reference Scenario or as an alternative scenario. Drilling of wells directly into the repository could be considered as an alternative scenario.

4.2.2 Development of interaction matrices for the Reference Scenario

The Process System to be described in the interaction matrix contains near-field, far-field and biosphere FEPs. Instead of compiling all information in one large matrix it may be wise to split the matrix into sub-matrices, e.g. one near-field, one far-field and one biosphere matrix or, alternatively, include the far-field in either the near-field or biosphere matrix. This will make it possible to develop matrices with high enough resolution without being too large to be practical to work with. This also means that parts of the work with the development of the matrices can be done in parallel by sub-groups of the project members.

The following actions would be needed for developing interaction matrices for the Reference Scenario.

1. The project group together defines the system to be included in the matrix and the initial and boundary conditions for the Reference Scenario for which the matrices are to be developed. In addition, the boundaries between the different sub-matrices are defined, and general criteria for definition of the diagonal elements in the matrices are specified. Furthermore, the priority scale to be used for judging the importance of interactions in the matrices is defined. The definition of the Reference Scenario and the priority scale are properly documented.

2. The diagonal elements of the matrix are defined and the properties of each diagonal element specified. Once this is done, interactions between all diagonal elements in the matrices are identified and described. All diagonal element and interaction descriptions are properly documented. This part of the work can be made together by the whole project group, but it is probably more cost and time effective to split the group into smaller groups, where each group is working with one sub-matrix. However, the identification of interactions occurring over the boundaries between the sub-matrices should preferable be done by the whole project group together.

3. The contents of the matrices are audited against different lists of FEPs. 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.

4. The importance of all identified interactions in the matrices is judged for the initial and boundary conditions of the Reference Scenario using the pre-defined priority scale. The setting of priorities to interactions within each sub-matrix does not necessarily have to involve the whole project group, while it would be advantageous if the assessment of priorities of interactions over the boundaries of the sub-matrix could be made together by all members of the project group. The result of the prioritisation should be properly documented in terms of assigned priority and motivation to the assigned priority.

4.2.3 Development of interaction matrices for alternative scenario

In this proposed methodology, the intention is to use the interaction matrices developed for the Reference Scenario to study effects of different scenario initiating FEPs on the evolution of the Process System for the Reference Scenario. This could be done by identifying the diagonal element containing the properties that primarily are affected by a defined scenario initiating FEP, and then propagate the change of state of these properties through the matrix. This has not yet been fully tested in the earlier applications of the interaction matrix method in SKB studies. However, this approach was used in the SITE-94 project where the Process System was structured in a PID (Chapman et al., 1995), and it is strongly believed that a similar procedure can be used for a Process System structured in an interaction matrix.

To investigate the effects of scenario initiating FEPs on the Process System for the Reference Scenario the following actions are foreseen:

1. Possible scenarios/scenario initiating FEPs should be selected together by the project group, and the primary impact point of these EFEPs in the interaction matrices for the Reference Scenario should be identified. Existing FEP-lists can aid in the process of identifying the possible scenarios/scenario initiating FEPs. The selection process and descriptions of the selected scenario initiating FEPs (EFEPs) should be documented.

2. Once the primary impact points in the matrices are identified in terms of features/properties, the consequence of change of state of these properties are evaluated by going through all interactions directly affected by these properties. These interactions are found in the row of elements containing the diagonal element defining the affected properties. The questions that should be asked are: does this change in state of the property require a revision of the priority assigned for this interaction for the Reference Scenario, and, does this change in state of the property require the addition of additional interactions. If the answer is no to both two questions no further evaluation of the impact on properties affected by the interactions in these diagonal elements is required. If the answer is yes to any of these two questions, the properties affected by these revised interactions are the next impact points. These properties are found in the diagonal element in the column containing the revised or new interaction. The procedure is repeated for all pathways in the matrix until the answer is no to both of the above given questions. Changed priorities compared to the Reference Scenario should be documented and motivated, and the reasons for not changing the priority of interactions should also be given. New identified interactions should be described and a priority should be assigned and motivated.

When this procedure is completed an interaction matrix for an alternative scenario has been constructed where the pathways of revised and new interactions give the difference between this alternative scenario and the Reference Scenario.

4.2.4 Formulation of scenarios and calculation cases

The developed interaction matrices and the associated documentation can be used as checklists in the formulation of scenarios and calculation cases needed to analyse the scenarios. The translation of the information in the matrices to scenario descriptions and calculation cases will also expose areas where conceptual models, calculation models and data presently are lacking.

4.2.5 Documentation system

It is proposed that the documentation system developed in the previous applications of the interaction matrix method in SKB projects is used. This means that all matrix specific documents are compiled in a separate database with a unique name, while more general FEP descriptions are added to the already existing SKB FEP database.

4.2.6 Initial FEP-list

A first version of a FEP-list that can be used as a checklist in the construction of the interaction matrices has been compiled and is given as Attachment B to this appendix. This list is the original list of FEPs from the scenario development work in the previous safety assessment of SFR, completed with FEPs identified in the scenario construction work carried out as a part of the prestudy of the SFL 3-5 repository concept. The repository SFL 3-5 is intended for long-lived low and intermediate level waste and the barrier system is similar to that in SFR-1.

This version of the list is by no means complete. There exist a large number of international FEP-lists that should be used to audit the contents of the interaction matrices (see 4.2.2).

4.3 BENEFITS

Some advantages of applying the interaction matrix method with linked documentation in scenario construction and consequence analyses in general and in project SAFE are listed below.

- It is a structured method for identification of FEPs affecting the behaviour of a system. The simple matrix structure allows many persons to participate in the development of the matrix. It also forces the people involved to seek for interactions between all identified features/properties of the system in a systematic way. This together with the documentation requirements probably increases the possibility of identifying all FEPs relevant for the system.

- The fact that all persons involved in project SAFE can participate in the construction of the interaction matrix will probably increase the participants understanding what regards the overall behaviour of the system and the need for specific information.

- All matrix information, the content of the matrix and all decisions made during the scenario construction work is compiled in a database format and linked to the matrix. This makes the matrix and its content easy to review and re-evaluate.

- Applying a systematic scenario construction method for the first time has been shown to be resource intensive. One reason to this is that it general involves development of procedures and documentation systems. Using an already tested methodology with developed procedures and documentation systems should therefore be more efficient and less resource requiring.

- Applying the interaction matrix method in project SAFE could contribute to further refinements of the procedure and the documentation system. In addition, it will contribute in the building of the general SKB FEP database.

5 SUMMARY AND CONCLUDING REMARKS

This appendix gives a short description of the scenario methodology adopted in the previous safety assessment of SFR. Since then new methodologies for developing structured descriptions of how processes and interactions between processes affect the evolution of a repository system. Two such methods are briefly described. These methods are very similar, but they differ in the way the system is graphically structured. One of the methods is based on Process Influence Diagrams, PID, and the other on Interaction matrices.

It is proposed that the method based on Interaction matrices is used for the scenario work in project SAFE. The main reason for this is that the method already has been applied by SKB, which means that it will be possible to use already existing procedures and documentation systems. The proposed procedure involves the development of Interaction matrices for a defined Reference scenario and the use of these matrices to illustrate the effect of different Scenario initiating FEPs. The proposed procedure is described in Chapter 4 of this Appendix.

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Attachment A: Vocabulary

Assessment basis
All factors that should be considered in determining the scope of the analysis; these may include factors related to regulatory requirements, definition of desired calculation end-points and requirements in a particular phase of assessment. (NEA, 1996)

Disposal system
The physical extent of the system which needs to be considered in order to meet the assessment basis.

System components
All physical components of the system that by evolution in time could change states and therefore should be included in a description of the performance of the system.

Process System
The organised assembly of all phenomena (FEP) required for description of the performance of the system components and radionuclide behaviour in the repository system (Andersson et al., 1989)

FEPs
Features, Events and Processes that could, directly or indirectly, influence the release and transport of radionuclides from a repository. (Andersson et al., 1989)

Internal FEPs
FEPs belonging to the Process System.

EFEPs (External FEPs)
FEPs acting outside the Process System.

Scenario initiating FEPs and Scenario
Scenario initiating FEPs are one or a combination of EFEPs that by acting on the Process System influences the evolution of the Process System. The EFEP and the consequential development of the Process System constitutes a

scenario, i.e. an illustration of a possible future evolution of the Process System for a given set of initial and boundary conditions. (Chapman et al., 1995).

Scenario methodology

The methodology used to select and describe scenarios to be considered in a safety assessment. The methodology suggested by Chapman et al., 1995 and Skagius and Wiborgh, 1994, includes:

- a description of the disposal system including a definition of the system boundaries and the physical components of the system
- an identification and structuring of FEPs belonging to the Process System in order to obtain a systematic description of the Process System
- a list of EFEPs that might influence the Process System and their relevance of occurrence
- a list of chosen scenario initiating EFEPs and the motivation for the decision
- a description of how expert judgement has been used in the scenario methodology process.

Scenario analysis

The actual analysis of a chosen scenario. This includes a description of the scenario (evolution of the Process System for the scenario initiating EFEP), the conceptual and numerical models used and the results from the quantitative analyses. A scenario analysis should also involve an uncertainty analysis. (Skagius and Wiborgh, 1994).

Reference Scenario

A scenario chosen for comparison reasons. It should not be seen as the most probable scenario. A Reference Scenario is often a simplified scenario that is rather easy to define, e.g. it could be defined as the evolution of the Process System without considering the impact of scenario initiating EFEPs. (Skagius and Wiborgh, 1994).

Reference Case

In the quantitative analysis of the Reference Scenario a Reference Case could be selected for comparison reasons. The Reference Case does not necessarily include all processes and mechanisms described in the Reference Scenario. (Skagius and Wiborgh, 1994).

System uncertainty

The uncertainty associated with the identification, structuring and ranking of FEPs in the Process System. FEPs may be missing, their interdependences may be improperly described or missing, or the importance of interdependences may be misjudged. (Chapman et al., 1995).

Scenario uncertainty

The uncertainty related to the EFEPs in terms of the completeness problem (have all relevant EFEPs been identified) and the quantification of the impact on the Process System, time of occurrence, frequency, combination of EFEPs, or other relevant measures of the identified EFEPs. However, as scenarios should be seen as largely illustrations of possible evolutions of the Process System, a certain amount of uncertainty in these areas is quite acceptable. (Chapman et al., 1995).

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Attachment B: List of FEPs

In the following initial list of identified FEPs, the phenomena are grouped according to the different barriers in SFR. This grouping is not strictly correct since a large number of phenomena could belong to several groups, which not always is indicated. Since no screening has been made, the list also contains FEPs that essentially are similar.

The groups defined are:

1. Waste and waste matrix
2. Waste containers
3. Back-filled porous concrete
4. Other concrete structures
5. Clay barrier
6. Near-field rock
7. Far-field rock
8. Biosphere

FEP's not belonging to any of these groups are assigned to the group:

9. Others

1. Waste and waste matrix

Variety of waste types
Radioactive decay
Radiolysis
Ageing and degradation of cement
Cement/concrete leaching
Alkali-silica reactions
Calcite/brucite precipitation
Degradation of bitumen
Metal corrosion
Gas formation
Corrosion products
Degradation of ion-exchange resins
Radionuclide solubility
Degradation of organic material
Swelling of ion-exchange resins
Swelling of bitumen
Volume expansion of cement
Ettringite formation
Formation of Friedel´s salt
Complexing agents

Amines
Microbes
Diffusion
Advection
Sorption
Formation of colloids
Release of nuclides from waste matrix
Isotopic dilution
Attenuation

2. Containers

Corrosion of steel container
Gas formation
Corrosion products
Corrosive agents
Local corrosion, pitting corrosion
Effects of concrete on steel corrosion
Stress corrosion
Creeping of steel container
Cracking along welds
Cracking due to external mechanical load
Hydrostatical pressure on containers
Cracking of container due to inside pressure build-up
Initially defect containers
Ageing and degradation of concrete containers
Cement/concrete leaching
Alkali-silica reactions
Calcite/brucite precipitation
Expansion of concrete containers
Ettringite formation
Formation of Friedel´s salt
Internal voids
Diffusion
Advection
Sorption
Attenuation
Release of nuclides from containers

3. Back-filled porous concrete

Concrete quality
Un-filled spaces left after casting
Inhomogeneity
Blocking due to chemical reactions
Cement/concrete leaching
Alkali-silica reactions

Calcite/brucite precipitation
Ettringite formation
Formation of Friedel´s salt
Gas flow and transport
Water displacement
Non-capillary bound water
Cracking/fracturing due to high pressures
Steel doors in silo compartment walls
Degradation of cellulose
Complexing agents
Sorption
Diffusion
Advection
Release of nuclides from porous concrete

4. Other concrete structures

Ageing and degradation of concrete
Chemical interaction with components from bentonite
Corrosion of reinforcement
Gas formation
Corrosion products
Cracking due to internal pressure
Internal mechanical stressing
Outer mechanical stressing
Hydrostatic pressure
Swelling pressure of bentonite
Movements of rock blocks
Silo collapse
Erosion
Interaction between concrete and water
Cement/concrete leaching
Alkali-silica reactions
Calcite/brucite precipitation
Ettringite formation
Formation of Friedel´s salt
Isotopic dilution
Complexing agents
Microbial activity
Sorption
Diffusion
Advection
Blocking of gas release devices
Gas transport in gas release devices
Chock front in pH for incoming water
Displacement of water
Changes in properties of material in gas release devices
Release of nuclides from concrete structures and gas release devices

5. Clay buffer

Chemical degradation
Ion-exchange
Hydroxide attack
Swelling and density reduction
Initial density and inhomogeneity
Coagulation
Swelling into fractures in rock and in silo
Dilution due to collapse of concrete structures in silo
Mechanical stressing caused by volume expansion of silo
Erosion
Clack valve function
Gas channels
Water channels
Drying up due to gas transport
Resaturation
Water flow
Gas flow and transport
Diffusion
Anion exclusion
Surface diffusion
Sorption
Complexing agents
Colloid source
Release of nuclides from clay barrier

6. Near-field rock

Shotcrete with draining pipes
Nuclide transfer resistance, engineered barrier-rock
Skin zone effects
Nuclide transfer resistance, engineered barrier-top void
Gas penetration into rock
Non-sealed shafts
Degradation of sealing material
Rock creep
Alteration/weathering of flow paths
Degradation of organics
Degradation of rock bolts
Mixing/dilution
Erosion
High pH-plume
Water flow
Gas flow and transport
Diffusion

Anion exclusion
Surface diffusion
Sorption
Complexing agents
Colloid source

7. Far-field rock

Groundwater flow
Changes in groundwater flow
Changes in hydraulic conductivity
Channeling flow
Dispersion
Dilution
Water flow between repository vaults
Non-discovered fracture zones
Gas transport
Oxidising conditions
Colloids
Complexing agents
Sorption
Matrix diffusion
Changes in groundwater chemistry, saline-fresh
Dissolution and precipitation
Weathering of flow paths
Hydrochemistry, chemicals from the surface
Isotopic dilution
Changes in rock properties
Fracturing
Land uplift
Future drill holes/wells
Human activities that affect groundwater recharge
Changes in sea level
Re-use of drill holes
Climatic changes caused by human activities
Transition from sea to lake or river

8. Biosphere

Erosion of sediment
Accumulation in sediment
Accumulation in peat
Intrusion into accumulation zone
Isotopic dilution
Human induced changes in surface hydrology
Human induced changes in surface water chemistry

9. Others

Re-saturation after repository closure
Earthquake
Retrieval of waste
Underground dwellings
Arceological intrusion
Explosion caused by sabotage
Monitoring after closure
Loss of records
Geothermal energy production
City above the repository

Project SAFE – Prestudy

Appendix A3:

Near-field

Marie Wiborgh
Kristina Skagius

Kemakta

Contents

1 INTRODUCTION

1.1 BACKGROUND

The Swedish Final Repository for Radioactive Waste, SFR, is a repository for low and intermediate level waste. The original operational permit was granted 1988 and was complemented with an operational permit for regular disposal of waste in the Silo and subsequent grouting around the waste in 1992. In the operational permit for SFR-1 the authorities requested that updated analyses of environmental consequences should be performed within periods of ten years.

1.2 AIM OF PRESENT STUDY

The aim of the project SAFE is to carry out the requested revised safety analysis for SFR. The "Prestudy" is the first step of the project, with the aim to identify improvements that can be made and included in the revised safety analysis. The project SAFE is divided into the following subject areas:
- Inventory
- Near-field
- Far-field
- Biosphere
- Operation
- Design/closure
- Scenarios.

In this appendix improvements or necessary revisions related to the performance and the analysis of the near-field are identified. The other areas are treated in separate appendices in this report.

1.3 OUTLINE OF APPENDIX

This appendix is intended to cover topics related to the performance of the near-field. However, many of the above listed areas have a direct influence on the near-field analyses. Therefore comments are given to some topics that are discussed in more detail in the other appendices.

A general overview of the repository design and a description of the engineered barriers for the Silo and the four caverns are given in Chapter 2.

The major assumptions going into the analysis of the near-field in the final safety report [SSR, 1987] and the deepened safety report [FSA, 1991] are discussed in Chapter 3. Assumed initial near-field conditions and considered processes and properties related to the barrier performance and radionuclide release that may have to be revised in the new analyses are identified.

Topics identified by the regulatory authorities as important areas or areas that should be continuously updated when more information is available are identified in Chapter 4.

Finally, general topics of importance and needs of improvements are given in Chapter 5.

2 DESCRIPTION OF THE SFR FACILITY

2.1 GENERAL

The repository is situated in crystalline rock between 50 and 150 meters below the seabed. The water depth above the repository is about 5 m. The first stage of the repository SFR-1, which is in operation, consists of four caverns and one Silo together with a system of transport tunnels, see Figure 2-1.

At the end of 1996 a total volume of about 20 000 m^3 of waste have been deposited in SFR-1. The total amount of waste planned to be disposed of in SFR-1 after future extension is 90 000 m^3, of which 60 000 m^3 is planned to be stored in the repository parts described below.

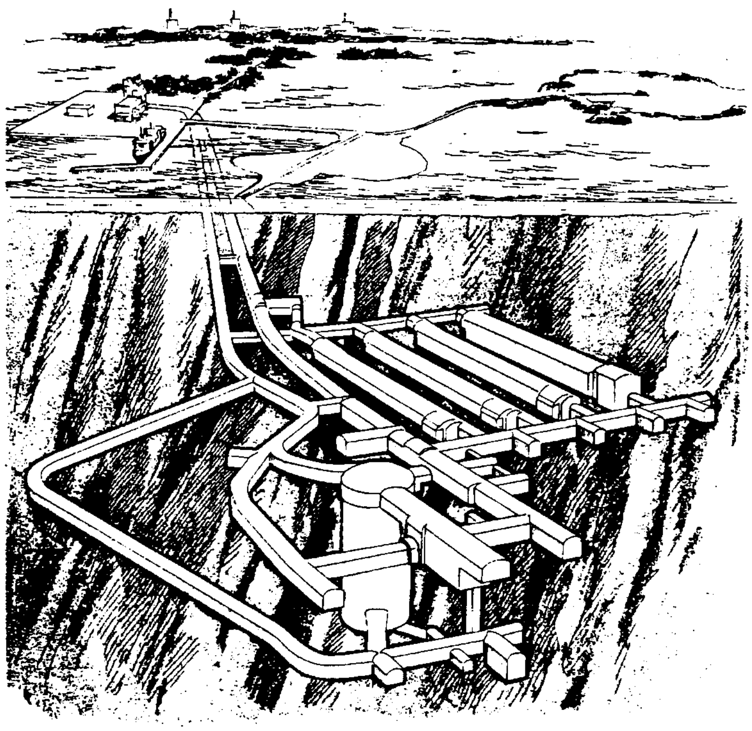

Figure 2-1. Overview of the SFR-1 facility.

2.2 SILO - BARRIER DESCRIPTION

The waste that contains the major amount of the activity will be stored in the Silo. The waste comprises mainly ion-exchange resins solidified with bitumen or cement but also small amounts of cement conditioned trash and scrap. The waste is packed in concrete and steel moulds or in steel drums. The Silo is designed to store 18 500 m^3 of waste. A schematic illustration of the Silo barriers is given in Figure 2-2.

The cavern is 70 m high and 30 m in diameter. The Silo is a vertical cylinder made of 0.8 m thick reinforced concrete. The height of the concrete Silo is 50 m and the inner diameter is 25 m. Between the concrete structure outer walls and the rock is a 1.2 m thick bentonite barrier. The bottom of the Silo is made of 1 m thick reinforced concrete, below which is a 1.5 m thick sand/bentonite (90/10) layer. The interior of the Silo is divided into vertical cells, 2.5 m square, by 0.2 m thick reinforced concrete walls. The waste is placed in the cells, which later are filled with porous concrete in campaigns.

The concept for closure of the Silo consists of a 1 m thick reinforced concrete lid on top of a sand barrier. The lid is perforated by sand filled gas evacuation pipes. On top of the lid is a sand layer and above that a 1.5 m thick layer of sand/bentonite mixture (90/10). In addition, the top part will be thoroughly backfilled with crushed rock and cement stabilised sand.

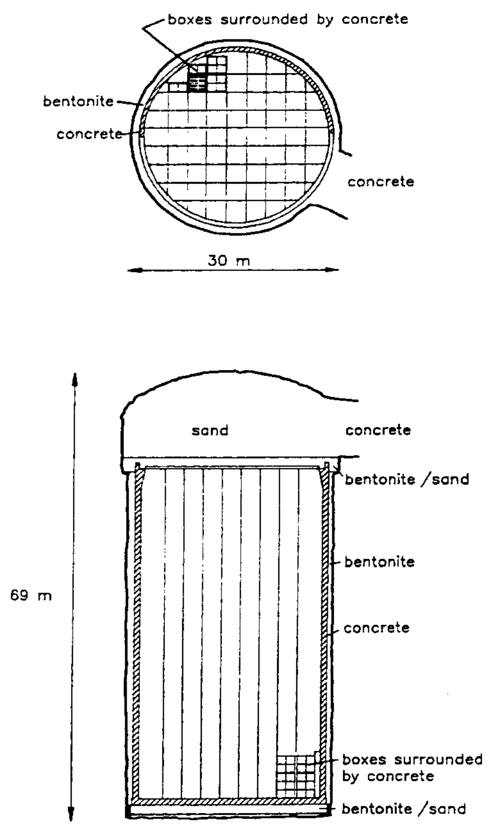

Figure 2-2. Schematic illustration of the Silo barriers.

2.3 CAVERN – BARRIER DESCRIPTION

SFR-1 consists of four caverns that are designed to host different types of waste. Schematic illustrations of the barriers in the cavern for intermediate level waste (BMA), the two caverns mainly for concrete tanks (BTF) and the cavern for low level waste (BLA) are given in Figure 2-3.

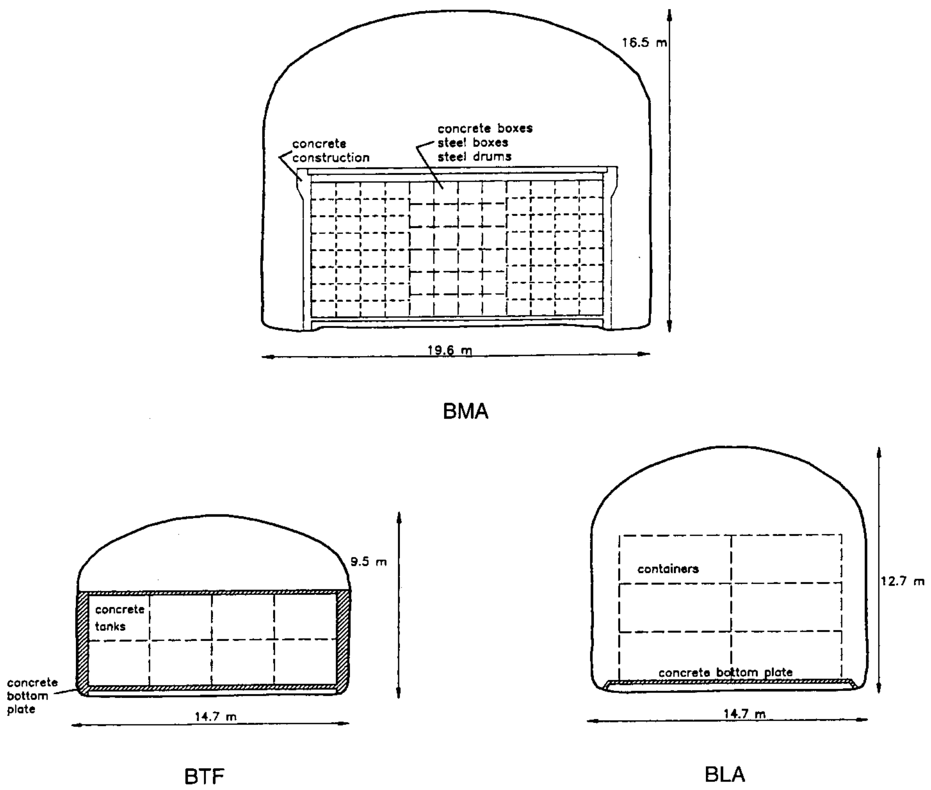

Figure 2-3. Schematic illustrations of barriers in BMA, BTF and BLA.

2.3.1 BMA- barrier description

In BMA medium level waste solidified in cement or bitumen and packed in steel drums, steel moulds or concrete moulds is disposed.

The cavern is designed to store 13 400 m^3 of waste and the total excavated rock volume is about 48 000 m^3. The cavern is 160 m long, 19.6 m wide and 16.5 m high. The cavern is divided into thirteen large and two small compartments. The thickness of the reinforced concrete walls are 0.4 m. The waste packages are loaded on a 0.3 m thick concrete floor in such away that the moulds make support for a prefabricated concrete lid which is put on top of each compartment when fully loaded. The tightness and the mechanical strength of the lid is further increased by pouring a 0.4 m thick concrete layer on top of the prefabricated lid.

At closure concrete plugs will seal the cavern. The 2 m wide space between the concrete structure and the rock wall is foreseen to be filled with sand. The void, above the compartment, is not foreseen to be filled.

2.3.2 BTF - barrier description

In BTF dewatered ion-exchange resins in concrete tanks, with an inner rubber lining, and smaller amounts of ashes in steel drums, with inner drums surrounded by concrete, are disposed.

BTF comprises two rock caverns, 1BTF and 2BTF. Each cavern is designed to store 7 900 m^3 of waste and the excavated rock volume is about 20 000 m^3. The caverns are 160 m long, 14.7 m wide and 9.5 m high. The rock walls are covered with 5-10 cm shotcrete. The waste is placed on a concrete floor with a thickness of 0.3 m. The concrete tanks are stored four abreast and in two layers. The waste packages are successively surrounded by concrete and finally a concrete lid is put on top of the waste packages as radiation protection.

At closure the space between the concrete tanks and the rock wall is foreseen to be backfilled with cement stabilised sand. The empty space above the concrete lid is not foreseen to be backfilled before the closure of the caverns with concrete plugs.

2.3.3 BLA - barrier description

In BLA non-solidified low level waste packed in standard ISO containers is disposed. The cavern is designed to contain 11 500 m^3 of waste and the total excavated rock volume is about 17 000 m^3. The cavern is 160 m long, 14.7 m wide and 12.7 m high. The rock walls are covered with 5-10 cm shotcrete. The containers are stored in three layers, two abreast on a 0.3 m thick concrete floor.

In BLA no additional barriers or filling materials are foreseen except concrete plugs to seal the cavern.

3 PREVIOUS WORK

3.1 GENERAL

In earlier assessments two time periods are distinguished, the "Saltwater period" and the "Inland period". During the saltwater period, up to 1 000 years after repository closure, the hydraulic conditions are assumed to be constant and the radionuclides are assumed to be directly released to the Baltic Sea. After 1 000 years the shoreline is assumed to pass the repository and the inland period starts. The hydraulic flow is assumed to increase momentarily and hereafter successively until a maximum is reached at 2 500 years. For the inland period the radionuclides are assumed to be released to a lake and/or a well.

In the final safety report [SSR, 1987] radionuclide release calculations for each repository part were reported for the saltwater period and the inland period. In the deepened safety analysis [FSA, 1991] new calculations on the release from the Silo during the saltwater 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 caverns 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 [FSA, 1991]. 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.

3.2 CONSIDERED INITIAL NEAR-FIELD CONDITIONS

After repository closure the pumping of drainage water will stop. Groundwater will start to infiltrate and the repository will become water saturated. In the assessment both the caverns and the Silo are assumed to be initially saturated at repository closure and the radionuclides are immediately dissolved. Reducing conditions are assumed initially to prevail and the geochemical conditions are determined by the large amounts of concrete in the repository.

Based on site investigations and modelling efforts the water flow has been estimated. The water flow rate in the disturbed area around the Silo was estimated to two times the flow rate in the undisturbed rock. The water flow in the caverns was estimated from a mass balance calculation on water entering and leaving the cavern part of SFR-1. The individual flows in the different caverns were estimated by proportioning the total flow with the tunnel bottom areas.

The initial activity content in the different repository parts is based on plans and prognoses from 1987, see Appendix A1 - Inventory. In the assessment it is assumed that the activity within each repository part is evenly distributed in the waste. At the start of the inland period, 1000 years after repository closure, no radionuclides are assumed to have been released and the activity content in the repository is only compensated for radioactive decay during 1000 years.

3.3 REPOSITORY NEAR-FIELD BARRIER PERFORMANCE

The assumed performance of the near-field barriers in former assessments is directly related to the division into a saltwater period and an inland period. In the assessment the barrier properties and the geochemical conditions in the near-field are assumed to be altered momentarily from one period to another.

The environmental conditions in the repository will influence the mobility of the radionuclides in the waste as well as the transport of elements in the barriers. Many investigations and research efforts on chemical and physical processes form the base for the predictions of the long-term performance of the engineered barriers.

Concrete barriers are present in several of the repository parts and the impact of mechanical and chemical degradation of concrete barriers was evaluated. A large number of chemical, physical and mechanical processes were identified that could lead to fracture formation within a time-scale of thousands of years and eventually to total mechanical degradation. In addition, an attempt was made to quantify the effects of these on the hydraulic properties of the concrete barriers. The result showed that fractures every 10 cm that penetrates half the thickness of the barrier only have a marginal effect on the hydraulic conductivity of an intact concrete as long as the fracture aperture is not larger than 1 mm. When fully penetrating, the effect of large fractures is drastic, while fractures with an aperture less than 10-3 mm has a negligible effect on the overall conductivity.

The main chemical degradation process was identified to be the leaching of components in the concrete since it affects pH and ionic strength in the system, and these parameters are important for radionuclide solubility and sorption. During early time periods, the alkali hydroxides in the concrete will give rise to a pH > 12.5 in the concrete water. Due to successive leaching of the alkali hydroxides the pH will gradually be decreased to about 12.5 and remain constant as long as free portlandite is present in the system. Depletion of portlandite will be followed by dissolution of the calcium silicate hydrate (CSH) gel in the concrete, slowly decreasing the pH from 12.5 to about 11. When the CSH gel is dissolved, the concrete is degraded leaving behind some brucite, calcite, low solubility silicates and the ballast material. At this stage the pH may still be as high as 10.4 due to dissolution of brucite and calcite. Based on this leaching sequence three sets of sorption coefficients (Kd:s) were defined, one for "fresh" concrete with pH > 12.5 and high ionic strength, one for "leached" concrete with pH > 11 and lower ionic strength, and one for "degraded" concrete with pH > 10.4 and lower ionic strength.

Leaching of the components of the concrete also affects the porosity of the concrete barriers. An evaluation of this effect revealed that the increase in porosity due to leaching of portlandite is negligible, while leaching of the CSH gel may result in a porosity increase from 10-15% to 22-27%.

The waste properties and the engineered barriers in the different repository parts differ. Therefore the evaluation of near-field barrier performance and estimated radionuclide releases are treated separately in the assessment. In the sections below an overview is given of assumptions made and processes considered in the evaluation of the function of the different repository parts.

3.4 SILO

3.4.1 Silo barrier performance

Intermediate level waste with the main part of the activity destined for SFR will be deposited in the Silo. The waste is conditioned in cement or bitumen in concrete or steel moulds and in steel drums. Porous concrete, concrete walls and a bentonite barrier surround the waste packages.

Degradation of the concrete barriers was divided into mechanical degradation and chemical degradation. As mentioned above, it could not be ruled out that fracture formation would occur within a time-scale of thousands of years. However, chemical depletion was estimated to take very long time. The reason to this is that the bentonite buffer, surrounding the concrete barriers, acts as a hydraulic barrier and thereby limits the rate with which components are leached and transported away from the concrete. The time for leaching of free portlandite from the concrete walls, bottom and lid was estimated to be of the order of thousands of years, while leaching of portlandite from the Silo interior would require several tens of thousands of years. The time required for complete chemical degradation of the concrete in the Silo walls, bottom and lid was estimated to be of the order of 200 000 years. The slow chemical degradation will maintain a high pH for long times, and this is of importance for sorption.

The effect of complexing agents on sorption was considered. Complexing agents present in the waste, such as EDTA, were assessed to be of negligible importance due to small amounts and low effect. Complexing agents from bitumen in an alkaline environment were also ruled out based on observations of a natural bitumen occurrence in highly alkaline groundwater in Jordan. Alkaline degradation of ion-exchange resins and cellulose are other sources of complexing agents. Experimental data available at the time of the safety analysis revealed that the alkaline degradation products of cellulose, especially isosaccarinic acid (ISA) could have large effect on radionuclide solubility in a concrete environment, but also on sorption onto concrete. However, no solubility limits were accounted for in the Silo, and the expected amount of cellulose in the Silo was estimated to be too low to have any effect on sorption.

Components leached from the concrete may affect the properties of the bentonite barrier, increasing the hydraulic conductivity of the barrier. An evaluation of the interaction between concrete and bentonite using simple mass balance and transport calculations and conservative assumptions regarding the leachability of the concrete, indicated alteration/degradation depths of the order of 0.5 and 0.3 m after 1000 years in the bentonite and in the concrete walls, respectively.

Gas can be generated in the Silo, mainly because of anaerobic corrosion of the steel present. Radiolytic decomposition and microbial degradation of organic materials are other gas generating processes. It was assumed that the generated gas would create gas channels through the porous concrete in the Silo to reach the gas release devices in the top of the Silo. For further gas escape out through the top bed of bentonite/sand a gas pressure gradient over the bentonite/sand must be established. Experiments had shown that this pressure gradient is of the order of 15 kPa for bentonite/sand and of the order of 50 kPa for pure bentonite. Based on these experimental data it was conservatively assumed that a pressure gradient of 50 kPa is needed to open up gas escape paths in the bentonite/sand in the top of the Silo. Both the creation of gas channels and the accumulation of gas and gas pressure build-up beneath the bentonite/sand will expel water from the interior of the Silo. The amount of water displaced depends on the capillary forces in the materials in the interior, i.e. on the size of pores and voids in the Silo interior. Experiments on porous concrete considered as backfill in the previous safety analysis showed that water expelled to form gas channels in the material was about 2% of the porosity. The same value was assumed for the concrete walls in the waste packages. The volume of water expelled due to the over-pressure of 50 kPa in the Silo was estimated based on the assumption that all water in voids in the waste packages and in gaps between packages are free water, i.e. not retained by capillary forces. The amount of free water in the waste cement mixture in the waste packages was assumed to be 30%, while 10% of the porewater in the porous concrete was assumed to be free water. It was further assumed that the amount of free water in the concrete walls, bottom and lid is negligible (capillary forces of the order of 1-2 MPa).

3.4.2 Silo radionuclide transport

In the radionuclide release calculations, the transport resistance in waste matrix, in waste packages and in the porous concrete was neglected and the system inside the concrete walls, bottom and lid was modelled as a mixed tank. The Silo interior was assumed to be initially fully water saturated and all radionuclides immediately dissolved in the water. No solubility limits of radionuclides were accounted for, and the pore volume and sorption on concrete packaging determine the concentration in the water in the interior of the Silo. During the first 1000 years, the "saltwater period", the hydraulic conductivity of the bentonite barriers was assumed to be low enough for making radionuclide transport by groundwater flow through the Silo negligible compared to transport by diffusion. The initial, gas induced, displacement of water and dissolved radionuclides from the Silo interior was considered by assuming plug flow of this water and dissolved radionuclides into the concrete walls and bottom of the Silo and, depending on the water volume, further out to the surrounding bentonite barriers. Radionuclide sorption onto the concrete and bentonite in the Silo walls and bottom was considered, both during the initial water displacement phase and during the subsequent transport by diffusion. The sorption data selected corresponded to the lowest value of those defined for "fresh" and "leached" concrete (see above). A conservative approach was also applied when selecting diffusivity and sorption data for the bentonite barriers.

Due to the potential degradation of the bentonite and concrete barriers during the first 1000 years, radionuclide transport was assumed to occur by both diffusion and groundwater flow through the Silo for year 1000 and onwards, the inland period. The water flow through the Silo was assumed to be of the same magnitude as in the surrounding rock with a vertical, downward direction. Due to land uplift and increasing hydraulic gradient, the water flow was assumed to increase proportionally with time resulting in a three times higher water flow at year 2500. Due to the topography of the area, land uplift after year 2500 was not expected to result in further changes of the hydraulic gradient. Consequently, the water flow through the Silo was assumed to remain constant after year 2500. The interior of the Silo was considered in the same way as during the first 1000 years with the exception that sorption data representative for "degraded" concrete were selected for all nuclides but cesium and strontium. These nuclides are expected to have lower sorption ability at higher pH and ionic strength and therefore data for "fresh" concrete was conservatively selected. For the bentonite barriers, the same sorption and diffusivity data as for the first 1000 years was selected.

Uncertainties in the long-term behaviour of the near-field barriers were studied by making different assumptions regarding the performance of the barriers and parameter variations. Consequences of deviations in the expected mechanisms for gas release and water displacement during the saltwater period were studied for the following assumptions:

- Not only the free water in the porous concrete and in the concrete walls of the Silo and the waste packages are expelled through the gas evacuation pipes when gas are creating transport paths inside the Silo but also the free water inside the waste packages. This increases the initial water volume expelled through the gas evacuation pipes with a factor of about 8.

- The gas release devices in the top of the Silo becomes blocked due to e.g. precipitation of calcite. As a consequence the gas pressure in the Silo interior must increase to the design pressure of the concrete structures before fractures are created in the upper part of the concrete walls and gas is released. The design pressure was assumed to be 280 kPa, i.e. about 5 times higher gas pressure is required for gas escape from the Silo, and about 5 times larger water volume is expelled compared to the case with gas release through the gas evacuation pipes.

- A reduction of the permeability or total blocking of the porous concrete by e.g. precipitation of calcite. This could result in pressure increase in both gas and water phase with fracturing of the concrete walls of the Silo as a consequence. The amount of water expelled depends on the maximum internal gas pressure and the extent and location of the permeability reduction.

- Fracturing of the walls and bottom of the Silo caused by ettringite formation or swelling of ion-exchange resins. Gas induced expulsion of water occurs through these fractures and the diffusive transport of radionuclides through the walls and bottom becomes faster because of these fractures.

- A high conductive fracture is present initially in the concrete bottom of the Silo. The build up of gas pressure in the Silo interior expels water from the interior out through this fracture in the bottom. The maximum internal gas pressure is determined by the pressure required to open up gas transport paths in the sand/bentonite in the top or by the design pressure of the concrete structure (blocked gas release devices).

For the inland period the consequence of restricted gas release from the Silo was addressed by assuming that the gas release devices are blocked at the start of the inland period at year 1000 and that a high conductive fracture is present in the concrete bottom of the Silo.

3.5 BMA

3.5.1 BMA barrier performance

The waste and waste packages in BMA are of the same type as in the Silo, but the activity content is lower. The main difference in design concerns the cavern contra Silo geometry and the sand/gravel backfill outside the walls and the unfilled void above the lid of the concrete compartments instead of bentonite barriers.

The groundwater inflow to the BMA at the start of the saltwater period was estimated based on an estimate of the groundwater flow in the whole repository area and the bottom area of the BMA cavern. This water flow into the cavern was assumed to prevail for the whole saltwater period. At year 1000, the onset of the inland period, it is assumed that the groundwater flow in the repository area and consequently the inflow to the cavern is increased with a factor of 10 as a consequence of land uplift. For the inland period a linear increase in the water flow with time was assumed until the year 2500, which is the time when no further impact on the hydraulic gradient from land uplift was expected.

The time required for leaching of the total amount of free portlandite in the concrete structures was estimated to be 1100 years assuming that all groundwater entering the cavern is flowing through the concrete compartments. The corresponding time for leaching of the CSH gel was estimated to be about 8000 years.

The consequences of complexing agents on sorption was considered and it was found that the amount of cellulose present, and potential formation of ISA, would require a reduction in the sorption coefficients (Kd:s) for some of the nuclides on concrete. Based on the information available at the time of the analysis a reduction factor of 50 was assumed for sorption of americium, curium, plutonium and technetium on concrete.

Gas can be generated in the cavern, mainly as a consequence of anaerobic corrosion and microbial degradation of organic materials. However, no effects of gas generation and transport on the release of radionuclides was considered since it was assumed that the concrete structures in the cavern initially contains fractures through which the gas can escape.

3.5.2 BMA radionuclide transport

In the analyses of the radionuclide release during the saltwater period the concrete compartments were assumed to contain fractures and have no resistance to flow. The groundwater entering the cavern was assumed to flow through the concrete structure evenly distributed in an upward direction. The transport resistance for radionuclides in the compartments was neglected and the system inside the compartments was modelled as a mixed tank. The radionuclides were assumed to be immediately dissolved, and sorbed to cement waste matrices in the waste packages. Potential sorption on concrete packagings was neglected. Sorption on internal compartment walls was considered as a transient process determined by diffusion into these walls. Radionuclides are released through the concrete lid by the water flowing through the compartments and by diffusion through the compartment walls. Before further transport into the rock, the radionuclides are mixed in the water in the void above the concrete lid.

For the inland period 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 the entire cavern was modelled as a well-stirred tank.

The first model was based on the assumption that despite degradation of the concrete structures and concrete waste packages most of the water flow in the cavern would take place in the voids outside the concrete structures. The flow distribution in the cavern was estimated by assigning a hydraulic conductivity of 10^{-2} m/s to the concrete structures. According to a theoretical evaluation, this conductivity would be of the same magnitude as the conductivity in concrete containing fully penetrating fractures with an aperture of 1 mm every 10 cm. Further assuming a hydraulic conductivity of 1 m/s in the void outside the concrete structures indicated that 4% of the total flow in the cavern would flow through the concrete structures. In the model it was further assumed that the radionuclides are evenly distributed in the concrete structures and the system inside the structures, and that the initial concentration is determined by the porewater volume and by sorption on cement and concrete. Radioactive decay during 1000 years was considered, but not any release of radionuclides during the saltwater period. A horizontal water flow through the cavern was assumed where 4% of the total flow passes through the concrete structures. Radionuclides are released from the concrete structures and the system inside by this flow and by diffusion in the direction of the groundwater flow. A vertical, upward directed release by diffusion to the water flowing in the void above the concrete structures was also included in the model.

In the second model the entire cavern was modelled as a well-stirred tank The initial concentration of radionuclides was assumed to be determined by the water volume in the cavern and by sorption on cement and concrete in the cavern. Radioactive decay during 1000 years was considered, but not any release of radionuclides during the saltwater period. Radionuclides are released from the cavern by the groundwater flowing through the cavern.

In both modelling approaches, the effects of concrete leaching on pH and ionic strength in the BMA interior was considered in selecting sorption data for concrete. In addition, to investigate the effect of a lower sorption due to the presence of ISA also during the inland period, the selected reduction factors were applied to the sorption coefficients defined for "fresh" concrete.

3.6 BTF

3.6.1 BTF barrier performance

The major waste type in these caverns was considered to be dewatered condensate resins in concrete tanks. The concrete tanks are surrounded by concrete backfill. For the first hundred years of the saltwater period, the good quality of the concrete tank walls, the inner rubber lining and the concrete backfill surrounding the tanks are assumed to prevent water flow through the tanks. The existence of small amounts of ashes in steel drums in BTF was not considered in the assessment.

The potential amounts of gas formed by corrosion and microbial degradation in the cavern can be expected to be released to the surrounding rock without any effects on the radionuclide transport.

3.6.2 BTF radionuclide transport

For the saltwater period it was assumed that the groundwater flow through the cavern 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 cavern was modelled as a well-stirred tank and the initial concentration of radionuclides was estimated from the activity and the volume of water in the tank. For the first 100 years after repository closure it was assumed that the concrete tank walls with inner rubber lining and the concrete would prevent water flow through the tanks and that diffusion through and sorption in the tank walls determined the radionuclide release to the rock. For the time period after 100 years it was assumed that the concrete backfill and the tank walls no longer prevent water flow through the tanks. Radionuclides in the tanks are released through the lid of the tanks by groundwater flowing through the tanks and through the walls of the tanks by diffusion. Sorption in the lid and walls of the tanks was considered and it was assumed that mixing occurs in the cavern void above the tanks before radionuclides are released to the surrounding rock. The initial activity inside the tanks was compensated for 100 years of decay, but not for any release during the first 100 years. In addition, the effect of groundwater flow through the tanks from the time of repository closure was also studied.

For the inland period the entire cavern was modelled as a well-stirred tank where the initial concentration of radionuclides is determined by the water volume in the cavern and by sorption in the concrete tanks and in the concrete backfill. Radioactive decay during 1000 years was considered, but not any release of radionuclides during the saltwater period. Radionuclides are released from the cavern by the groundwater flowing through the cavern. The effects of concrete leaching on pH and ionic strength in the BTF was considered in selecting sorption data for concrete. In addition, the amount of concrete available for sorption was reduced based on the estimated increase in porosity due to leaching.

3.7 BLA

3.7.1 BLA barrier performance

In BLA the waste is mainly stored directly in containers. No transport resistance or sorption on waste and repository materials are accounted for in the safety assessment.

Gas generated in BLA due to anaerobic corrosion of metals and microbial degradation is expected to be transported away without affecting the radionuclide release.

3.7.2 BLA radionuclide transport

In the radionuclide transport calculations for the saltwater period it was assumed that the groundwater flow through the cavern is upward directed and evenly distributed and that the magnitude is determined by the groundwater flow in the rock. The whole cavern 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 cavern. The initial concentration of radionuclides in the cavern was estimated from the initial activity and the porewater volume in the cavern.

The same model as above was used for the inland period. Any sorption on materials in the cavern was neglected. Only radioactive decay during 1000 years was considered, but not any release of radionuclides during the saltwater period. Radionuclides are released from the cavern by the groundwater flowing through the cavern.

4 PREVIOUS REGULATORY REVIEWS

In this chapter topics addressed in the review by the regulatory authorities [SKI/SSI, 1992] of the deepened safety assessment [FSA, 1991] as important areas and/or areas that must be continuously updated are summarised. Here mainly topics related to the performance of the near-field are given.

4.1 RADIONUCLIDE INVENTORY AND WASTE

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

Complexing agents in the repository can increase the mobility of radionuclides and thereby increase the radionuclide release from the repository. The reviewers request 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.

In the review the importance of controlling materials and substances in the waste that may influence the long-term performance is identified. Areas of interest are especially:
- microbial degradation of organic material in the waste and related gas formation
- swelling of bitumen conditioned ion exchange resins
- complexing agents in waste
- sulphate and sulphate sources in the waste influencing ettringite formation).

4.2 HYDROGEOLOGY

Inland period
The reviewers identify that the performed estimates of the direction of the groundwater flow through the caverns are probably true but based on very crude modelling. A remark is also given on the groundwater flow increase when the sediments are not considered.

Wells
The reviewers comment 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.

4.3 ENGINEERED BARRIER PERFORMANCE

Concrete in structures
The reviewers think that there are large uncertainties in estimates of the mechanical properties of concrete in structures after 1 000 years. However, only marginal influence is expected on the radionuclide release.

Porous concrete backfilling
It is doubtful if SKB has been able to show that the porous concrete has the ability to transport gas for long-term periods.

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

Swelling
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.

Better understanding of complexing agents
The degradation of organic material in the waste and the formation of complexing agents (e.g. ISA) may lead to decreased sorption. The reviewers say 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.

5 IDENTIFIED NEEDS OF IMPROVEMENTS

The identification of improvements is mainly focused on assumptions made in former analyses and remarks given by the reviewers on the SFR safety assessment. It is very likely, that the foreseen scenario work on the performance of the near-field barriers, where all aspects will be worked through in a more systematic way, may display additional improvements or new topics of importance for the safety assessment, see Appendix A2 - Scenarios.

In this chapter topics where improvements can be made are identified. A division is made into three main areas:
- Processes
- Data revision
- Near-field modelling.

5.1 PROCESSES

Identified areas where better understanding of the processes can be of potentially importance for assumptions going into the near-field analysis are listed below together with their possible effects:

Near-field hydrology

- The updated structural models for the SFR repository and subsequent revised near-field hydrology modelling may result in new estimates on the amount and the direction of groundwater flow around the Silo and in the different caverns, see Appendix A4 - Hydrology. In former assessment the division of the groundwater flow between the caverns were based on the cavern bottom areas. A change in amounts and directions of groundwater flows will have a direct influence on radionuclide releases and may also influence engineered barrier degradation processes.

- Plugging of caverns and tunnels and modifications of cavern barrier design, for example backfilling of empty voids, may influence the magnitude and division of water flow between BMA, 1BTF, 2BTF and BLA. In addition, it may influence the division of water flow in the interior of the different caverns. Changes in water flow will have a direct influence on near-field barrier performance, see above.

Near-field barrier performance

- In the release calculations the repository is assumed to be initially saturated. Evaluation of the time for Silo saturation may delay the time for and the magnitude of anaerobic corrosion and gas generation. In addition, the availability of radionuclides in the waste may be influenced.

- The saturation of the Silo and the surrounding buffer may influence the integrity of the Silo concrete structures. Causes may be uneven water uptake and development of varying swelling pressures in the bentonite buffer.

- In former assessments the activity is assumed to be evenly distributed in the waste in the different repository parts. Therefore, will the concentration of the radionuclides not exceed any solubility limits and all radionuclides must be assumed to be initially dissolved in the porewater. The availability of some radionuclides may be determined by their solubility in the waste matrix, if waste type specific considerations were taken into account.

- The engineered barriers in the Silo were in the release calculations considered to restrict the radionuclide transport and/or to have a sorption capacity. The bentonite surrounding the mantel restricted the water flow through the Silo structure for the salt-water period (the first 1000 years) whereas for longer periods the buffer and the Silo concrete structure were assumed to have no limiting effect on the water flow. The degradation of the surrounding bentonite buffer, the sand/bentonite mixtures in the bottom and at the top and the concrete structure should be updated with newer findings and research results. This may give arguments for assuming a restricted water flow through the Silo structure for longer time periods.

- In the safety assessment of the caverns generally very conservative assumptions were made regarding the hydraulic properties and the sorption capacity of the concrete barriers. Revaluation of the concrete degradation process may provide arguments for applying less conservative assumptions regarding these properties.

- The initial properties of the porous concrete and the predictions on long term performance of the backfill are essential for the evaluation of the gas transport capacity and the potential amount of contaminated water displaced from the Silo. A revaluation of these properties should be made considering the actual composition of the porous concrete and utilising information from the operation of the Silo.

- In the assessment no credit has been taken to the retardation in sand/gravel backfill and the near-field rock. To be able to do this, 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.

- 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 and colloid formation at the front of a potential pH-plume.

- The increased amounts of bitumen must be considered in the new assessment. Estimates of the time for and the extent of the water uptake in bitumen may limit the radionuclide release. The potential swelling of bitumen conditioned waste may also influence the barrier integrity.

- The possibility of taking credit of the available sorption capacity in the waste and barrier materials in BLA could be investigated. The effect of backfilling the tunnel with for example sand (if possible from mechanical point view) may also be worthwhile to study.

- In former assessment all waste in BTF was assumed to be dewatered ion exchange resins in concrete tanks. The long-term behaviour of the increased amount of waste packages with ashes in BTF must be investigated, e.g. potential for gas formation.

- Possible mechanisms for obtaining radionuclides in gaseous form should be investigated, e.g. formation of gaseous C-14 by microbial degradation of organic materials.

5.2 DATA REVISION

The data used in the former safety assessment must be updated with recent findings from ongoing research and investigations. In addition, data to be used in the safety assessment on waste allocation and waste composition may have to be modified based on information on already stored waste in SFR or changed future plans. Useful information can also be gained from the SFR control program on repository conditions. At last, the aim must be that used data are consistent with data used in other safety assessments in SKB for the same type of materials and conditions.

Identified areas, where new information can be or are available, are listed below together with their possible effects. No efforts are made here to evaluate their relative importance for the safety assessment.

- Information available from the SFR control program on groundwater chemistry. Major changes in ground water composition will have an influence on assumptions regarding the environmental conditions in the repository and thereby also on concrete degradation and bentonite degradation processes. In addition, processes such as; corrosion, microbial degradation and colloid formation may be influenced.

- An important source can be information available from the SFR control program on water uptake in the bentonite buffer surrounding the Silo. An evaluation of data based on real conditions may indicate that unsaturated conditions in bentonite buffer and the interior of the Silo must be considered for longer time periods.

- Information in existing waste register and future plans on waste production may reveal changes in waste inventory, waste conditioning procedures and waste allocation that may influence assumptions made on the amount and the availability of radionuclides in the different repository parts. For example, the activity is foreseen to decrease in the Silo whereas the activity increases in BMA and the amount of waste conditioned with bitumen is increasing.

- In the safety assessment work for the SFL 3-5 investigations have been made on element solubility limits in concrete influenced repository environment. The use of these data on solubility limits may limit the source term concentration of some radionuclides, if the activity is not assumed to be evenly distributed between different waste types.

- There exist more measurements and evaluations of sorption, diffusivities and permeabilities for concrete materials, bentonite and rock materials than in the former assessment. In the new assessment the aim must be to have a consistent use of data compared to similar SKB assessments.

- More information is available on existing strong complexing agents, the formation of complexing agents and their effects on sorption. The use of this information may result in a decrease/increase of sorption in individual waste types or repository parts.

- The gas formation in the repository is foreseen to be dominated by corrosion of waste packagings and reinforcement bars in concrete structures. Radiolytic decomposition and microbial degradation is foreseen to be of minor importance. The gas formation rates must be updated with respect to changes in waste composition and the use of alternative packagings. In addition, the gas release and the influence on the radionuclide release will have to be studied.

5.3 NEAR-FIELD MODELLING

The data revision and the revaluation of processes of importance for predictions of long-term barrier performance will have an influence on the modelling efforts needed to estimate the radionuclide release from the different repository parts. Assumptions made on waste, initial conditions and barrier performance in the modelling of the radionuclide release from the near-field may have to be changed due to newer data and/or better understanding of important processes.

Below some examples are given on areas where a revision of the former near-field modelling may be required:

- In the former assessment the transport resistance in waste matrices was not considered but the sorption capacity of conditioning cement was taken into account. The increased amounts of bitumen conditioned waste imply that the source term modelling must be revised.

- To be able to make more realistic estimates on barrier degradation and radionuclide transport a better model is needed of the water flow within as well as between the different repository parts. This is also essential for estimates of chemical interactions between different repository parts.

- The importance for the radionuclide release during the inland period of a more realistic conversion from conditions assumed to prevail during the saltwater period may be worthwhile to study. The previously used approach where the conditions are assumed to momentarily change at the start of the inland period to conditions that should cover the entire inland period is a very conservative assumption.

- In previous assessment the activity content and the waste properties were averaged within the different repository parts. To be able to make waste type specific considerations, e.g. solubility limitations, more detailed near-field modelling is required.

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Project SAFE – Prestudy

Appendix A4:

Far-field

Sven Follin
Johan Andersson
Johan Holmén
Carl-Lennart Axelsson

Golder Associates AB

Contents

1 INTRODUCTION

This appendix aims at identifying the needs and to suggest a strategy for the geosphere analyses of the SFR in connection to the planned update of the performance assessment of the SFR within the framework of the SAFE-project. The appendix is however strongly focused on the hydrogeologic modelling since it turns out that the hydrogeology is the most important geosphere aspect to consider in the performance assessment. Furthermore, evaluation of the geologic model of the site is presented elsewhere (Axelsson and Hansen, 1997).

2 PREVIOUS WORK

The main hydrogeological modelling of the SFR performed previously was made in preparation for the application for an operating permit submitted in 1987 (SKB, 1987). Some additional modelling was done in preparation for the in-depth safety assessment (SKB, 1991) submitted to SKI and SSI in August 1991. In 1993 SKB included the SKB (1991) results in an update of the 1987 safety assessment (SKB, 1993). Regarding evaluation of groundwater chemistry and gas transport in the geosphere the conclusions made in SKB (1987) were repeated in the updated assessment.

2.1 HYDROGEOLOGIC MODELLING

2.1.1 Three-dimensional finite element modelling

The hydrogeological modelling of the groundwater conditions at the SFR site made by Carlsson *et al.* (1987) assume a homogeneous fluid, steady-state flow and a simplistic structural geology. The rock is described as a porous medium. Structures (fracture zones) were modelled by assigning the decided hydraulic conductivities to the finite elements representing the structures. Modelling was done in three dimensions, but with rather coarse meshes.

Domains and scales

The simulations were carried out on three scales:

- regional scale (~ 85 km^2, element size ~ 300 m) ,
- local scale (~ 2 km^2, element size ~ 150 m), and
- repository component scale (< 1 km^2, element size ~ 50 m).

A nested modelling approach was applied, such that the large-scale models provided input to the smaller scale models.

The regional model simulations were used to describe the influence of regional lineaments on the groundwater conditions around SFR. It also provided boundary conditions for the local model. The Singö zone represents the western boundary. It could be noted that the horizontal zone H2 is extended right to this boundary in the model, although in the final structural model presented by SKB (1993) the H2 zone does not extend that far. However, a variation case with a less extensive H2 zone was also included in the analysis.

The local model simulations were used to describe the effect of local lineaments on the groundwater conditions around SFR. Furthermore the hydrogeological input to the safety assessment in SFR was derived from this model. The access and construction tunnels were modelled as a single pipe. The repository vaults were modelled as a single 10 m thick horizontal slab ("flake") (i.e. as single room) and the silo as a rectangular block.

In the repository component model an attempt was made to actually discretise each individual vault (in fact, two different repository component models were set up, but this is not of a major relevance here). However, the model was only used to estimate the inflow during the operational phase to the different parts of the repository and the resultant changes to the potentiometric head field of the surrounding rock under the present day draining stage. Boundary conditions were obtained from the local model for the case with a "drained" (i.e. pumped see below) repository.

Cases and boundary conditions

The regional and the local model analysed the "sub-marine" (called the "saltwater" period in most SFR-documents) period and the "inland" period as two different steady-state simulation cases. For the inland period boundary conditions were altered in order to represent the effect of the land rise.

For the sub-marine period an "excess head" of 0.5 m was applied as a boundary condition at the Singö zone (in the regional model), motivated by an interpreted excess head based on extrapolations of measurements at the H2 zone before construction of SFR. This boundary condition transmits into the local model and is, in fact, the only driving force for groundwater flow in the simulations of the marine period. Given the measurement conditions and the added complexity of saline water, the precision in the interpretation may in fact be quite uncertain, making the boundary condition questionable.

The local model also evaluated a case without a proper seal in the access tunnel (only for the marine period) and a case with a drained repository. The latter case was used to provide boundary conditions for the repository component model.

2.1.2 Stylised evaluation of changing boundary conditions

In preparation for the in depth safety assessment (SKB, 1991) Axelsson *et al.* (1991) analysed some aspects of land rise and its effect on the dilution in a domestic well. These calculations were carried out in two-dimensional cross sections 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. That is, 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 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 to the boundary conditions) would be several thousands years.

2.2 MEANS OF CALIBRATING AND VERIFYING MODELS

A check of the total inflow to the repository in the repository component model was the main entity used in calibrating and verifying the hydrogeological modelling. The simulated values were compared to measured values and a fair match between simulated inflow and measured inflow was obtained. It was however shown that the inflow was quite sensitive to assumptions on the skin in the rock/vault interface. There was also an attempt made to compare calculated and measured hydraulic heads in two boreholes, but this comparison was quite inconclusive. Carlsson *et al.*, (1987) did not report a value of the estimated inflow in the local scale model.

Several attempts were made to see whether the regional model could reproduce the excess head measured during the pre-investigations. For the analyses made it seems very difficult to reproduce this head, and it appears more likely that there was no flow (or a very limited one) under the sea floor before constructing the SFR facility. In fact, given the uncertainty in the interpretations of the field measurements (see 2.1.1), such a conclusion would lie within the error bars.

A clear indication that the fresh/salt water interface is not at steady-state (i.e. a qualitative support for the model by Axelsson *et al.*, 1991) is that the (present) depth to the interface, as observed in the existing coastal wells, is much less than that calculated assuming steady-state flow (see Axelsson, 1986).

2.3 USE OF HYDROGEOLOGICAL MODELLING IN SAFETY ASSESSMENT

The hydrogeological input to the safety assessment of SFR (SKB, 1987 and 1991) was taken from the local scale model. Parameters were provided for the marine (saltwater) case and the inland case. A more detailed description of the near-field and migration evaluations is given in the near-field and migration appendices respectively.

2.3.1 The silo

The release model used for the silo utilises the darcy velocity in the rock surrounding the silo. This darcy velocity was taken directly from the simulation model. This flow was used directly for diffusion cases and doubled for cases assuming flow through the silo, thereby approximately accounting for the loss of flow resistance from the silo. (Doubling the flow is correct for a cavity in a homogeneous porous medium, but only approximate in a heterogeneous medium).

2.3.2 The rock vaults (BMA, BTF and BLA)

The flow through the rock vaults were estimated from the "turnover" in the "flake" representing the rock vaults in the local scale models. The model resolution did not allow for a direct division of the flow through the "flake" on the individual vaults. Instead the assumptions was made to divide the flow in proportion to the bottom area of each vault. This assumption is more relevant for the saltwater period, where the simulation resulted in a vertical flow, than for the inland case where the (simulated) flow is more horizontal. Furthermore, the assumption is based on the existence of a homogeneously fractured rock mass, which could be questioned.

During the inland period an attempt was made, in the 1991 assessment, to estimate the division of the through the waste concrete compartments and the flow through the fairly unfilled space above these compartments in the BMA. Assuming horizontal flow and a much higher permeability of the space surrounding the waste compartments led to a conclusion that only 4% of the water through flow actually pass the waste.

2.3.3 Retardation

The potential reduction of impacts through retardation in the rock was discussed, but was not considered an important safety barrier. For the saltwater case the simulated migration path is too short to produce any retardation. For the inland case a one-dimensional streamtube model, which depends on groundwater flow and flow wetted surface was used (Moreno *et al.*, 1987). A more thorough description of the migration modelling is given in Appendix A5 Radionuclide transport.

2.4 GAS MIGRATION IN THE GEOSPHERE

Thunvik and Braester (1986) estimated the gas transport capacity of the rock. They concluded that the transport capacity very significantly exceeds the potential gas production from the waste.

2.5 GROUNDWATER CHEMISTRY

The chemical composition of the groundwater at SFR, based on groundwater sampling made during drilling and in specially designed boreholes, is provided in the safety assessment report itself (SKB 1987, and SKB, 1993). It is concluded that the repository itself, due to it iron and concrete content would control the chemistry in the vaults. The future composition of the groundwater in the geosphere will depend on the large-scale ground water flow regime.

3 PREVIOUS REGULATORY REVIEWS

3.1 HYDROGEOLOGY

The main regulatory review of the hydrogeological modelling of SFR was made by SKI in connection to the evaluation of the operating permit (SKI, 1988 and Andersson and Gustafsson, 1988). Additional review comments were made in the joint SKI/SSI review of the in-depth safety assessment (SKI, 1992).

In general, SKI and SSI considered the description on geology, topography and land uplift of the repository area as satisfactory, but several points were raised.

- The reviewers 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. Variation cases set up by the reviewers assumed uncertainties of one order of magnitude.
- Andersson and Gustafsson (1988) 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 used in the modelling, this was due to the limited computer capacity available at the time of simulation.
- It was noted that the assumed excess head during the marine period might be incorrect. It may be more probable to assume almost zero flow during marine period.
- The interaction between land rise, saltwater and groundwater movement is only qualitatively described by SKB. It was noted that if a quantitative study is undertaken it needs to incorporate the transient effect from the changing boundary conditions.
- SKI's own 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.
- The groundwater flow calculations of Carlsson et al., (1987) 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.

- The inventory of wells in the Forsmark area shows that the probability of finding a well in the area will be relatively high. Furthermore, the reviewers concluded that no or little dilution could be accounted for a well in or in close connection to the repository.

In support to the SKI review Andersson and Gustafsson (1988) performed a set of independent flow variation cases, although using the same finite element mesh as used by Carlsson et al., (1987). These simulations, in combination with the comments given above, led SKI and SSI to use approximately 4 times higher fluxes for the inland case compared to the SKB values.

3.2 GAS MIGRATION IN THE GEOSPHERE

The SKI review 1988 confirmed the SKB statements considering the insignificance of the gas migration in the geosphere. The SKI view was based on analyses by Pahwa (1988), a consultant independent from SKB. The issue was not brought up in the 1992 review.

3.3 GEOCHEMISTRY

Groundwater chemistry was not discussed in the SKI review 1988. In the SKI/SSI review 1992 concluded that apart from the BLA, the chemistry at SFR is dominated by concrete. They furthermore state that if the geochemistry had had a more decisive effect on the barrier function of the repository, site-specific data would have been necessary.

4 COMMENTS ON PREVIOUS WORK

4.1 HYDROGEOLOGY

4.1.1 Modelled processes

The approach of using nested modelling is attractive. In particular, it seems relevant to associate calibration to the repository component scale rather than to the local scale or the regional scale. However, the main hydrogeological assumptions, i.e., a homogeneous fluid density and steady-state flow, are potentially unsatisfactory. As pointed out by Axelsson *et al.* (1995) and Follin *et al.* (1996) the northern parts of Uppland were uplifted quite recently as a remaining effect of the latest glaciation. The current rate of the land rise in this part of Sweden is of the order of 0.006 m/year, which means that the marine period of the SFR site will cease in about 1000 years.

The study by Voss and Andersson (1993) clearly demonstrates that groundwater flow in a coastal system submitted to land rise is not in steady state, but in a transient mode due to the continuous change of recharge and discharge areas. This will have a significant effect on the flow paths in the system. Furthermore, the flow may be affected by the density effect of the salt water (the salinity at this shallow depth is about 1%). The effects of transient boundary conditions and the effect of variable density flow has previously (i.e. by Axelsson *et al.*, 1991) only been considered in a qualitative sense at the SFR facility.

4.1.2 Structural model

The underlying structural model for the hydrogeological modelling has recently been scrutinised by Axelsson and Hansen (1997). They suggest some modifications to the model used by Carlsson *et al.* (1987). It is yet to be decided whether further modifications are necessary.

Under all circumstances it is evident that a new hydrogeological model of the SFR site should be based on the most modern structural model of the site. Furthermore, outside the repository area the structural model is less certain. One of the criticisms of the old structural model was the inconsistency between the local and regional structural models. A new hydrogeological evaluation should explore the impact of these uncertainties - for example by variation cases.

4.1.3 Modelling approach

The are also several comments that could be made regarding the hydrogeological modelling of Carlsson *et al.*, (1987). In addition to the remarks already brought up in the reviews by SKI (see Section 3) the following can be noted:

- Modelling was made with the technology of the mid-1980's, which among other things means that the model discretisation did not allow for a description of individual rock vaults. (The repository was represented by a plate). This means that the old model cannot say anything about the distribution of flow in or between individual rock vaults.
- The inconsistency in mass balance, probably caused by too coarse numerical discretisation, decreases the credibility of the groundwater turnover estimates.
- The effects of the transient boundary conditions and the effect of variable density flow were not quantified.
- There were only limited attempts at estimating radionuclide migration properties.
- Uncertainties in the model assumptions, and the implications of the fact that crystalline fractured rock is heterogeneous, but modelled with a homogeneous porous medium model were not discussed thoroughly. The proper choice of conceptual model should be made from an assessment of the relevance of the heterogeneity of the rock in relation to the predictive needs.
- The effect of seals and how to place seals in a meaningful way was only partly addressed by a single calculation case.
- New developments of the modelling approach for near-field release, radionuclide transport and biosphere modelling may call for calculations of hydrogeological responses - not considered previously.

These points, as well as the SKI review issues need to be addressed in an update of the hydrogeological model of the SFR site.

4.2 GROUNDWATER CHEMISTRY

It appears 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 appendix on migration). 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.

4.3 GAS MIGRATION IN THE GEOSPHERE

The issue of gas migration in the geosphere should be reconsidered in a scenario and process analysis of SFR. It may also be worthwhile to scrutinise the previous assessment, but given the strong conclusions already made it appears that gas migration in the rock will still remain as a minor issue.

5 IDENTIFICATION OF OBJECTIVES FOR ADDITIONAL HYDROGEOLOGICAL MODELLING

A revised hydrogeological model of the SFR site should:

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

Evidently these main objectives are coupled. In arriving at these objectives it is also important to recognise that the identified needs for a hydrogeological input to the safety assessment have been revised through the SAFE-project.

5.1 CREDIBILITY AND UNDERSTANDING

The main objective of revising the previous structural and hydrogeological modelling is to demonstrate an understanding in the safety aspects of the rock's barrier performance at SFR. This means that models need to be credible. At a minimum the models should not be contradicted by available information. The hydrogeological model should strive for making a best estimate and consistent model of the safety relevant aspects of current and future groundwater flow at the site and evaluate uncertainties in these estimates.

5.1.1 Best estimate and consistency

The term "best estimate" does not have a statistical meaning. On the contrary, it is a quite judgmental concept. It is thought to mean the most representative model given our present understanding of the SFR site. "Best estimate" does not necessarily imply average values. In practice, the approach has to be to base the "best estimate" model on the actually available data and to strive for consistency in this model.

The most important component in making a hydrogeological modelling credible is consistency. Consistency calls for a rigorous quality assessment of the data and methods used, regardless of how flow and transport in the rock will be conceptualised. In addition, use of state-of-the-art presentation techniques, where the facility, structures and model results are visualised in three dimensions is an important means to achieve confidence in the modelling work undertaken.

5.1.2 Evaluation of uncertainty

Using so called "conservative assumptions" in hydrogeological modelling is not appropriate without an uncertainty analysis as it, indeed, is difficult to prove which assumptions that are conservative and which are not from a safety point of view. Thus, conservative assumptions made at the radionuclide release and transport calculations level should be based on the uncertainty estimates (at least a bracketing study) made from the hydrogeological modelling.

Possibly, alternative conceptual models need to be used in order to assess the uncertainties associated with processes, heterogeneity and lack of data. On the other hand it may turn out that a set of variation cases would be sufficient to handle the uncertainty estimates.

5.1.3 Use of calibration

As noted by Axelsson *et al.*, (1995) there exist data on hydraulic and chemical responses from the construction of the SFR facility and onwards. It needs to be realised that these data are of varying relevance and quality. Nevertheless, the data should be checked and used for model construction and parameter calibration whenever found practical and meaningful. Currently, the available database is brought together in a formal report, which also will include an assessment of the overall use of the data (Axelsson *et al.*, 1997).

Before discussing specific calibration objectives the following general remarks should be made:

- The need and accuracy of calibration depends on the purpose of the model being calibrated. Calibration first of all implies that the model parameters are adjusted such that simulated data reproduce measured data.
- Calibration of a model in a heterogeneous medium is a kind of scaling - the model parameters become equivalent values with respect to the target functions.
- In order to answer whether the calibration is relevant, one first needs to determine if the parameter being calibrated against is relevant or representative for the intended purpose of the model.

Given the complex history of the SFR site and the heterogeneity of the structural and hydrogeological conditions the above points imply that only a subset of the measured data are likely to be useful for calibration. The data most suitable for calibration appear to be (Axelsson *et al.*, 1997.):

- the inflow to different parts of the SFR facility, perhaps divided into a few distinct stages of the repository construction, and

- the time for breakthrough of salt-water from the Baltic Sea in different fracture zones.

More specific suggestions are given in Chapter 6 where various potential modelling approaches are discussed. Unfortunately, most of the groundwater pressure data gathered at the SFR site appear either to be irrelevant for the modelling problem ahead or too uncertain. Due to the difference in scale between the point measurements of hydraulic head and the scale of the suggested modelling approaches, combined with the fact that head measurements alone provide little information on conductive structures which are connected to the SFR facility, a calibration against head measurements from the control programme can only be recommended for some proper spatial averages of these measurements.

5.1.4 Change of boundary conditions

As already noted the ongoing land-rise after the latest glaciation will alter the hydrogeological boundary conditions at the SFR site during the "life time" of the radionuclides. In the shorter term the flow system will change once the pumping stops and the repository is closed just as the construction and dewatering of the repository once disturbed the natural groundwater flow. All these changes to the groundwater flow system should be considered, either by discussing their effect or by direct modelling.

5.2 EXTENDING AND SEALING THE REPOSITORY

At present it is undecided whether the vault system of the SFR facility will be extended and also if an evaluation of such extensions should be part of the SAFE-project. If it is decided to analyse an extended vault system, these changes should be included in the hydrogeological modelling. The decision for inclusion of such an analysis evidently lies with SKB.

The closure of the SFR facility means that the tunnels need to be sealed in order to diminish 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, however, not a part of the modelling.

5.3 INPUT TO SAFETY ASSESSMENT CALCULATIONS

There are different potential uses of the results of the hydrogeological modelling in the safety assessment, which puts different demands on the hydrogeological modelling.

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

Groundwater flow is an important input to the near-field source term calculation (see Appendix A3 Near-field). According to planned developments of the source term calculations 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, 2BTTF and BMA) - where the hydraulic conductivity of the different vaults may vary in time due to degradation,
- 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 5.1 (the calculated release of radionuclides would be significantly reduced if it possible to show that flow predominantly will take place in the buffer or else outside the waste form), and
- estimation of flow mixing between different vaults (this may affect the judgement of chemical interactions between the vaults).

It also appears to be important to visualise the 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 5.1) 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.

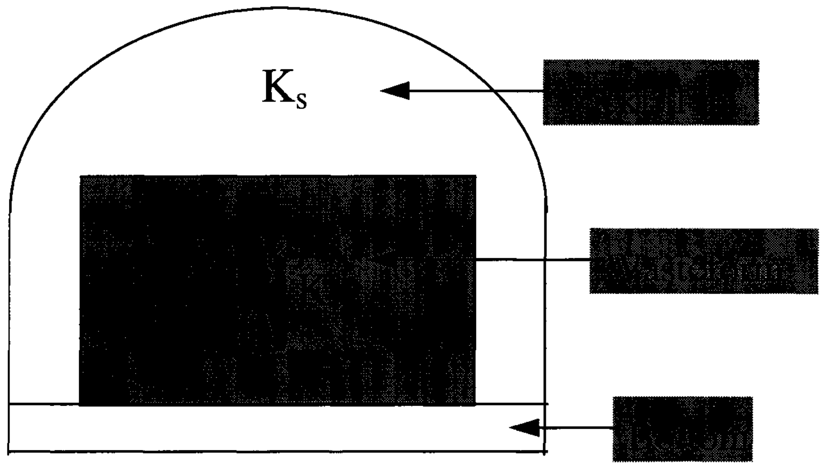

Figure 5.1: *Schematic cross-section through the BMA. For the estimation of radionuclide release it is very important to determine whether the groundwater flows through the waste form, the backfill, the bottom part or through the backfill. Flow directions are also very important.*

5.3.2 Input to migration calculations

For far-field migration the plan is to use transport models similar to the ones presently applied within SR 97 (i.e. FARF31). The following input would be needed:

- (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.,* 1997),
- "non-sorbing" breakthrough time,, and
- dispersion estimates.

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.

5.3.3 Input to biosphere assessment

The hydrogeological input for the biosphere modelling is the following:

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

It needs to be made clear that 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). It is the responsibility of the biosphere modelling to provide this information, but the responsibility of both to assure the consistency.

It has been suggested (see the appendix on the Biosphere) that the stratification of the sedimentary layers, which today constitute the sea floor, will be affected by the regression of the Baltic Sea. In principle this would alter the topography and should thus be considered in the hydrogeological modelling as well. The issue needs to be raised, but not necessarily analysed quantitatively. Evidently, the hydrogeological analysis would need to show that anticipated changes in the sea floor topography would not significantly change the conclusions about the groundwater flow at the repository.

6 SUGGESTED MODELLING

It is suggested that the hydrogeological modelling be 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.

The precise dimensions of the different scale models would be a modelling decision. Evidently the repository scale model would be embedded in the regional scale model and with boundary conditions taken from the regional scale model. However, the optimal placement of the repository scale boundaries could only be judged when the regional model results have been produced. In a similar fashion the migration scale model would be embedded in the regional scale model, but not necessarily in the repository scale model. Again it will be the results of the regional scale modelling that would identify where migration modelling would be necessary.

6.1 REGIONAL SCALE ANALYSIS

6.1.1 Objectives

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, and
- 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

- provide input to 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). For this latter reason it is suggested to carry out the regional scale analysis in two steps, where the first step should be designed to explore the need to include density dependent flow in further analyses.

6.1.2 Representation of rock and structures

On regional scale the only viable option to represent flow in fractured rock is to assume an equivalent porous medium intersected by major discontinuities. The assumption that flow in fractured rocks may be treated as an equivalent porous medium intersected by major discontinuities calls for an assessment of the uncertainties regarding the location and properties of the latter. In short, major alternative structural interpretations should be implemented and studied. Their impact on the groundwater flow conditions at the location of the SFR facility should be evaluated. The rock mass between large-scale discontinuities could either be represented as a homogeneous medium with carefully selected conductivity values, possibly different in different regions, or as a stochastic continuum, using blocks of a large scale. The level of discretisation would have to be adapted to the code and computer capabilities.

6.1.3 Representation of repository

The SFR facility should be modelled in a quite simplistic manner in order to obtain reasonable boundary conditions for the repository scale model.

6.1.4 Density effects

According to Freeze and Cherry (1979) the density effects could possibly be neglected for salinities less than 1% and such a conclusion can also find support from the study by Voss and Andersson (1993). In order to study the relevance of this statement for the problem of interest it is thus suggested:

- to formulate a fairly generic simulation case of sea-level change, but with salinities and land-rise adopted to the SFR,

- evaluate the resulting groundwater flow and migration of saltwater with a coupled transient density dependent flow and migration code by first performing simulations with the density effect included and then perform similar simulations but without the density variability,

- analyse the problem in three-dimensions (using simple geometry codes such as SWIFT or FEMWATER, or in two dimensions using, e.g., SUTRA (Voss, 1984), and
- compare the migration of seawater and the flow in the repository of the density dependent model with these quantities of the constant density model.

If these simulations show that density effects can be omitted, further evaluations will be made with versatile non-density dependent codes - such as the GEOAN-code, Holmén (1997). One should note, however, that the effect of a changing seawater level will still need to be analysed, but this can be accommodated within the suggested approach. On the other hand, if density effects are shown to be important the remaining option is to retain this effect at least in the regional scales.

More detailed and definitely three-dimensional modelling would be needed for evaluating the importance of uncertainty in regional structures and to produce time varying boundary conditions for the repository scale analysis. When transferring calculated head from a regional model to a local model, special care and consideration need to be taken, especially if the regional model includes a density dependent flow system.

6.1.5 Boundary conditions

The model domain should be large enough to be insensitive to the vertical and bottom boundary conditions. A proper size could possibly be established through the first set of density dependent simulations. More attention is needed for the top boundary. The following approach is suggested:

- The sea should be represented as specified pressure (or constant head for the constant density cases).
- Land rise is modelled by changing sea level and moving the seashore accordingly.
- Today's topography is easily available. Topography and seashore position in the past could be obtained from the hydrogeology evaluation within the Östhammar feasibility study (Follin *et al.*, 1996). Detailed sea-bottom surveys are available (Axelsson *et al.*, 1983). Regarding future topography and re-location of the sediment layer stratification co-ordination with the biosphere assessment would be needed.
- On land the boundary conditions should be given as precipitation rather than fixed head (see Holmén, 1997). This would allow calculation of

groundwater discharge / recharge including development of lakes etc. instead of postulating these important processes.

- There is no need to treat the unsaturated zone as long as the interest is the flow in the repository

See also 6.3.

6.1.6 Calibration

If possible, saline wells in the Well Archive at SGU will be used in a quantitative manner, e.g., to match a fresh water/sea water interface.

6.1.7 Output

The first stage of the regional scale analysis will determine whether density effects can be omitted and will thus define the scope for the remaining analyses. Consequently, the first step analyses should be performed at an early stage.

If density effects can be omitted the second step regional scale analysis will be used to define boundaries and boundary conditions for the repository scale, and possibly the migration scale analyses. If density effects cannot be omitted, the regional scale analysis will be used to search for periods where the repository scale and migration scale analysis can be applied without density effects for limited periods of time, and to provide proper boundary conditions for these periods. If needed these time periods will be co-ordinated with different time periods for the degradation of the repository barriers.

Under all circumstances, the regional scale analysis will be used to demonstrate and visualise the effect of seashore changes over time (past and future). In particular the analysis should provide elaborate arguments whether there is any reason to believe that groundwater flow could occur in the repository region during the marine period. As already noted, it appears most likely that such flow will not occur and that the measured excess head lies within the uncertainty range of the applied measurement technique and interpretation.

6.2 REPOSITORY SCALE

6.2.1 Objectives

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 (see 5.3.1),

- identify the portion of rock where geosphere 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, and
- assess the uncertainties in the predictions which are caused by the uncertainties in the adopted modelling approach,
- enhance the understanding by means of visualisation.

In meeting the specified 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.

6.2.2 Representation of structures and rock mass

In order to take full account of the rock heterogeneity the most suitable approach would be to represent the rock with a discrete fractured medium. This approach is clearly needed when assessing migration properties as migration concerns the scale of individual fractures (see further discussions in Section 6.3). However, in order to calculate fluxes to and within large tunnels and were precise assessment of the aerial distribution of these flows are not of primary interest, the discrete network approach may be too complicated (see e.g. Holmén, 1997) - in particular when considering the size of system that needs to be modelled simultaneously. For this reason the best advice for a versatile repository scale model would be to still opt for a porous medium description, but with close attention to formulating proper averages of the rock mass permeability, or include the rock mass heterogeneity by use of the stochastic continuum approach.

The most recent structural interpretation (see Chapter 4) should be implemented. Variation cases should be used to acknowledge uncertainties in the structural interpretation as well as the associated transmissivity distribution. However, as discussed by Axelsson et al., (1997) there will be no need for re-interpretation of the short-range hydraulic tests.

The rock mass between zones should be treated as a porous medium possibly with anisotropy. Hydraulic anisotropy has been found to be of great importance for the application of the porous medium approach to the groundwater conditions at the Äspö Hard Rock Laboratory (Rhén and Forsmark, 1996). In general, little or no information on anisotropy is available at the SFR site.

However, based on the experience from the Äspö HRL and in northern Sweden (Olsson, 1979) anisotropy assumptions could be based on information regarding fracture orientations and the stress field. Uncertainties could be addressed by variation cases.

The permeability of the sea bottom sediment will certainly affect the total inflow estimates, which will complicate calibration of the model. A reasonable range has to be tried. Furthermore, the stability of the bottom sediment could certainly be questioned, which means that there may be a need to define variation cases in order to show that performance assessment predictions are robust to assumptions regarding the bottom sediment.

6.2.3 Representation of the repository

The actual tunnel and rock vault system should be reproduced in the model. Given the size of the repository a realistic block size would be about 10 m. In such a case the vaults will be one block across. The division between different parts of the tunnel (c.f. Figure 5.1) need to be considered as well as the rock between vaults.

The silo should be represented by low permeability blocks. If it is believed that the silo will degrade it may be possible to increase this permeability after a certain time.

Plugs should be represented by blocks of low permeability placed at different sensible locations in the tunnels, such that the principle effects of plugs could be evaluated. It is, however, not suggested to evaluate how to actually achieve these plugs with available construction methods.

To avoid a model with a very fine and complicated mesh, one possibility is to use special elements in the model. Tunnel elements that contain element properties or local systems of smaller elements defining the very local properties of the tunnels and the silo can be applied (Holmén, 1997). In this way local properties, such as concrete constructions, containers and backfill materials etc. could be included in the model without the need of a very fine discretisation outside of the tunnels.

6.2.4 Density effects

Given that the regional scale analysis (see above) shows that there is no need for density dependent flow - or at least could be simplified by dividing the problem into suitable time stages - we suggest a fully transient but constant density and three-dimensional porous medium model approach, with special attention to tunnel flow. Such a model has been developed for SFL 3-5 (Holmén, 1997) and we suggest to build on this experience.

If the regional scale analyses shows that density dependent flow cannot be neglected, alternative models should be used (see 6.1.4). An alternative strategy then would need to be developed. However, the need for such a strategy is considered quite low.

6.2.5 Boundary conditions

Bottom and vertical boundary conditions should be taken from the regional scale model. When transferring calculated head from a regional model to a local model, special care and consideration need to be taken, especially if the regional model includes a density dependent flow system and represents a time dependent scenario.

The top boundary conditions will be produced in the same way as for the regional model (see 6.1).

The excavated repository could be represented by a zero pressure at walls of tunnel and vaults. It may be necessary to introduce a skin (which will further complicate calibration with total inflow).

The model could also evaluate the effect of wells and the possibilities for dilution in wells. The placement of wells should be co-ordinated with the biosphere and scenario analysis.

6.2.6 Calibration or model verification

Axelsson *et al.* (1997) currently explore which data would be suitable for calibration or model verification. In general, the following seems worth to calibrate on:

- total inflow before and after construction of the silo,
- inflow to the different tunnels and vaults before and after construction of the silo,
- time for breakthrough of Baltic Sea water in different fracture zones.

In principle all structural variants should be applied to the calibration data.

Clearly there exist much more data, including small-scale interference tests and inflow measurements during different phases of the construction work. However, the measurements are uncertain and would often require introduction of additional parameters, which quickly would mean a diminishing return of such exercises.

A problem for calibration is that many different parameters may be capable of explaining the observed phenomena. In particular both the bottom sediment permeability and the development (or not) of a skin would have a large impact both on inflow and the migration time for the seawater. This means that calibration could not be unique. It should rather be seen as a means of producing potentially viable models.

6.2.7 Output

The output of the repository scale analysis will provide quantitative information regarding:

- flow in the rock surrounding the vaults,
- flow through rock vaults and within the different parts of the vaults (see Figure 5.1),
- flow between rock vaults,
- mass balance over any selected parts of the model,
- impact of plugs, and
- streamlines (path lines if transient flow) from repository to the model boundary and wells, which would identify the region of interest for the migration scale modelling.

It is strongly suggested that the simulation results be visualised using three-dimensional visualisation tools.

6.3 MIGRATION SCALE

6.3.1 Objectives

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 5.3.2), and
- estimates of the distribution of biosphere release points (see 5.3.3).

(See also the appendices on migration and on the biosphere analysis).

The location and size of the DFN model will depend on the outcome of the regional and repository scale models. If groundwater flow is directed upwards it may be unnecessary to construct a migration scale model at all, since the overburden thickness is too small to motivate any significant retention.

6.3.2 Representation of structures and rock mass

When modelling migration through fractured crystalline rock the interaction between flow in the fractures and rock itself is of high significance. The heterogeneity of the rock mass affects radionuclide transport properties significantly. Furthermore, in assessing such transport the correlation between flow, flow wetted surface and migration paths must be considered (Andersson *et al.*, 1997).

Three different approaches for detailed groundwater flow modelling, stochastic continuum, discrete fracture network and channel network, are explored in the "alternative model project" (Ström and Selroos, 1997) within SR 97. The final selection of model concept to be used for the SFR assessment should await the outcome of this alternative model project. However, at least on paper the discrete fracture network (DFN) approach has some advantages over the other approaches:

- it actually tries to describe flow as it can be observed from individual bore holes and tunnels and thus lend itself to a co-ordinated interpretation of structure and hydraulic data,
- for transport applications it has the intuitive benefit of actually describing the interaction between flow and the rock itself, and can thus provide direct estimates of the correlation between flow and the flow wetted surface,
- it can be adapted to relatively large-scale applications (e.g. Geier, 1996), but then it is necessary to exclude the details of the fracture network (i.e. only include the large features) in the outer regions of the model domain.

The DFN-approach should be applied to calculate migration retention parameters, notably the F-parameter, in a format similar to what is presently tried within SR 97 (see Dershowitz *et al.*, 1997).

In the DFN approach one differentiates between deterministic and stochastic structures. Whereas the locations and properties of the former are more or less fixed, the locations and properties of the latter vary randomly in space. Hence, the mean behaviour and coefficient of variation needs to be studied by means of Monte Carlo simulations. The minimum structural and hydro-geological information required is:

- fracture orientations from outcrops, excavations and boreholes,
- fracture trace lengths from outcrop and excavations (to estimate fracture sizes)
- double packer hydraulic tests

For migration, crucial parameters are the flow distribution on the plane of the fractures and the conductive fracture frequency, which together with the Darcy velocity will determine the "F-ratio" (i.e. $a_r L/q$, see Andersson *et al.,* 1997). It is suggested that the approach taken for estimating these properties in SR-97 are applied here as well.

6.3.3 Representation of the repository

The actual tunnel and rock vault system will be positioned at a boundary in the model - and not represented explicitly. (See also 6.3.5).

6.3.4 Density effects

Given that the regional scale analysis shows that there is no need for density dependent flow - or at least could be simplified by dividing the problem into suitable time stages - we suggest a fully transient but constant density and three-dimensional discrete feature model approach.

6.3.5 Boundary conditions

The location and size of the DFN model will depend on the outcome of the regional and repository scale models. If groundwater flow is directed upwards it may be unnecessary to construct a migration scale model at all, since the overburden thickness is too small to motivate any significant retention. However, if the regional or repository scale analyses indicate a large region "down-stream" the repository, this region would be selected as the region for the migration scale analyses. Boundary conditions would be generated from the larger scale models, although attention would be required concerning flow consistency over this boundary.

6.3.6 Calibration

There appears to be little data suitable for calibration of a discrete network model (or any other detailed scale hydrogeological model). However, possibly some of the inflow interference tests could be evaluated within this approach.

6.3.7 Output

The DFN-approach should be used to provide estimates of the ratio between flow-wetted surface, migration path length and darcy velocity (i.e. the "F-ratio" a_rL/q), and the non-sorbing breakthrough time t_{tot} Andersson *et al.,* 1997). These parameters fully control the retardation in the rock and could also be used to derive the input parameters to the migration code FARF31.

7 CONCLUSIONS

This appendix has identified potential needs for updated hydrogeological modelling of the SFR in connection to the planned update of the performance assessment of the SFR within the framework of the SAFE-project. The objectives of such updated modelling should be to present a credible representation of the hydrogeological system, to explore effects of seals and repository extensions and to provide input to the release and transport calculations of the assessment. The last objective has led to the conclusion that an important focus of the modelling should be to determine the flow through the vaults under different conditions as this flow appear to be a very important quantity in the radionuclide release calculations.

The suggested modelling should use relevant data and apply modern modelling tools and techniques, but should be geared towards the objectives. For this reasons it is suggested to apply a set of complementary and sometimes nested approaches, where each model approach is set up in order to address a specific set of questions. Answering these questions would motivate simplifications made in subsequent steps of the modelling. For example, it is suggested to evaluate the effect of varying density in a simplified geometry, but to model the transient boundary conditions and the vault geometry in a realistic three-dimensional geometry. Such a simplification, and focusing of modelling efforts appear to be well suited for determination of flow through the rock vaults. However, for modelling migration in the rock mass a discrete network representation of the rock may be necessary, whereas the full geometry of the vaults may be less critical in this case.

To the extent possible the models should be compared with existing data on flow and Baltic water breakthrough. However, in making such comparisons the accuracy of the measurements and the precision of the models need to be considered. A one-to-one match cannot be expected.

It appears that careful geochemical evaluation of the site would only be necessary if more credit is placed on migration in the geosphere. If such an evaluation is considered it should be co-ordinated with the regional groundwater modelling.

The issue of gas production should be reconsidered in a scenario and process analysis of SFR. However, given the strong conclusions already made it appears that gas migration in the rock will still remain as a minor issue.

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Project SAFE – Prestudy

Appendix A5:

Radionuclide transport

Luis Moreno

Department of Chemical Engineering and Technology
Royal Institute of Technology

SUMMARY

A critical revision of the previous safety assessments (SSR and FSA) made by SKB on the Final Repository for Radioactive Operational Waste, SFR is presented. The review of the Deepened Safety Assessment (FSA) performed by SKI and SSI is also discussed. Based on this critical revision improvements are suggested. Hydrology, formation of complexes, and long-term behaviour of the barriers are some of the aspects where the safety assessment could be improved.

Contents

1 INTRODUCTION

The Final Repository for Radioactive Operational Waste, SFR, is located under the Baltic Sea close to Forsmark nuclear power plant. It is a repository for low and intermediate level waste. The first stage of SFR, which is in operation, comprises a disposal capacity for 60 000 m^3. A second stage with a capacity of approximately 30 000 m^3 is planned.

A Safety Assessment, SSR (SKB, 1987) was presented by SKB to the authorities in 1987. In 1992, a Deepened Safety Assessment, FSA, (SKB, 1991) was presented in accordance with the operating permit obtained in 1988. In that report, some areas are covered in detail, namely, the effects of gas production, the effect of complexing agents from the degradation of cellulose and the change in the hydrological regime due to land rise. The Deepened Safety Assessment (FSA) was reviewed by SKI and SSI and presented in two reports (SKI, 1992; SSI, 1992).

The aim of this report is a critical revision of the previous safety assessments and of the review of the Deepened Safety Assessment (FSA) made by SKI and SSI. From this critical revision improvements are suggested.

2 PREVIOUS WORK

In this section, a description of the previous Safety Assessments is presented. This is based on the Deepened Safety Assessment, FSA, presented in 1991 to the governmental authorities (SKB, 1991) and on the previous Safety Assessment presented in 1987, SSR (SKB, 1987).

The Swedish Nuclear Power Inspectorate (SKI) and the Institute for Radiation Protection (SSI) wrote a report (SSI-SKI 1992) about the evaluation of the Deepened Safety Assessment presented by SFR. A summary of this report is also presented.

2.1 PREVIOUS SAFETY ASSESSMENT

The SFR repository is located in bedrock 50 m below the Baltic Sea. Due to land rise, the Baltic Sea will move to the east and the rock above the repository will emerge from the sea. The land rise rate is currently about 6 mm/year, which means that at the same rate the repository will reach the coastline in 1000 years.

The land rise will modify the hydrology in the region where the repository is located. So, two different periods were distinguished. One is the salt-water period when the repository is below the Baltic Sea. Another is the inland period that starts when the repository reaches the coastline in 1000 years. Since the hydrology is very different between these periods, calculations were made for both periods.

During the salt-water period, the groundwater flow will be very small since no local gradients are created under the sea. The sea above the repository will be the primary recipient. As the transport distance from the repository to the recipient is short, the retardation in the far field was conservatively neglected.

During the inland period the repository will not be below the sea and the zone above the repository will be one of recharge. The hydraulic gradient and the groundwater flow will then be increased. The flow direction will be downward in the repository. Regarding the recipient, a small lake will be formed downstream of the repository and a well may be bored there after 2 500 years. In this period, when the recipient is a well or a lake situated some thousands of metres from the repository, retardation by diffusion and sorption in the rock matrix are considered.

A base scenario was defined for each part of the repository for each period. This corresponds to "the best estimate," i.e., the scenario that is expected to take place. Due to the large inaccuracies, some variation scenarios are defined, to study the release of radionuclides under extreme situations. The probability that one of these scenarios takes place is small. No calculations combining several of these extreme scenarios were therefore made.

2.1.1 The base scenario during the salt water period

During the salt-water period, the repository is below the sea bottom. The hydraulic gradient is small and the groundwater flow is horizontal or weakly upward. The primary recipient is the sea located above the repository. From scarce over-pressure measurements, the groundwater flow is estimated to be 0.2-0.5 $l/m^2 \cdot year$. The retardation in the geosphere is conservatively neglected since the distance from the repository to the recipient is short. Sorption on cement, concrete or filling material retards the radionuclide migration.

Silo. Radionuclides are transported from the silo by diffusion in the concrete walls and bentonite barrier. Water flow through the silo is negligible. It is assumed that the anaerobic corrosion of metals and its corresponding gas generation start successively in the silo. After some tens of years all the surfaces will be corroded.

The gas formed expels a certain volume of water to create channels for the gas escape to the upper part of the silo. Gas accumulates below the bentonite/sand barrier at the top. The pressure increases until channels in the bentonite/sand barrier open and gas escapes. Over-pressure in the silo pushes some water into the silo walls and bottom. Further radionuclide transport takes place by diffusion in the concrete walls and the clay barriers. When pressure equilibrium is reached, a steady state is obtained and no more water will be expelled from the silo.

BMA. The water flow takes place in BMA between the concrete building and the rock walls. It is determined mainly by the flow rate in the surrounding rock, since the flow resistance in the sand and the unfilled space above the building are small. A fraction of the water can flow through some fissures in the concrete building. Nuclides diffuse through the waste matrix, concrete barriers to the water flowing in the fissures or around the repository. Sorption can vary depending on the degradation level of cement and concrete and the existence of substances that can form complexes.

BTF. Water flows in BTF mainly in the unfilled space above the concrete tanks. A fraction of the water can flow through the concrete tanks. This flow will be increased by degradation of the concrete with time. Radionuclides emigrate with the water flowing in the tanks and by diffusion to the unfilled space. Sorption can be reduced by the presence of substances that can form complexes.

BLA. Water flow through BLA is determined mainly by the water flow in the rock surrounding the BLA and the possible influence of other caves. Radionuclides are dissolved in all the water contained in the BLA. The water flowing in BLA transports radionuclides out.

Radionuclide transport through the geosphere. Nuclides escaping from the different parts of the repository are transported with the groundwater through fractures and fracture zones into the recipient (Baltic Sea). Retardation caused by sorption on fracture surfaces and diffusion into the rock matrix are small due to the short travel times.

2.1.2 The variation cases during the salt water period

Several extreme cases were studied for the salt-water period. For the base scenario, the dose to the biosphere is dominated by the release from BMA. The release from the silo is negligible. Release from the silo could be increased if contaminated water was expelled from the silo. This could occur, for example, if gas passages are blocked.

If some passages for the gas transport are blocked, an additional volume of contaminated water could be pushed out from the silo. Therefore, if the silo walls or bottom are damaged, this water volume may be directly expelled from the silo. Two situations may be distinguished. The volume of contaminated water expelled is small and the water does not reach the rock. Further transport takes place by diffusion. In the other case, the volume of water is large and a certain volume of contaminated water is directly expelled into fractures in the rock. Several extreme cases, related to gas production, have been studied:

- Clogging of the evacuation pipes.
- Reduction of the porous concrete permeability.
- Clogging of the porous concrete at a certain level.
- Silo walls are cracked (e.g., by ettringite formation).
- A few large fractures appear on the silo walls.
- A combination of two cases: Clogging of the evacuation pipes and a fracture at the bottom.

No extreme cases have been calculated for the other parts of SFR (BMA, BTF and BLA).

2.1.3 The base scenario during the inland period

When the inland period starts, the zone above SFR will be a zone of recharge and the groundwater flow will be increased. After 1 000 years, the water flow rate is estimated to be 5 l/m^2year. After 2 500 years, the water flow will be 15 l/m^2year. The water flow will be downward during the inland period.

The hydraulic conductivity of the barriers will increase with time. This means that transport by flow will be the main mechanism for radionuclide migration from all parts of the repository, even from the silo.

Radionuclides that escape from the repository will be transported by the groundwater flowing in fractures and fracture zones. Since the transport distance will be long, the retardation caused by diffusion and sorption in the rock matrix must be considered in the calculations. Formation of complexes and channelling can reduce the effect of diffusion/sorption in the rock.

The radionuclide inventory at the start of the inland period is determined directly by the initial inventory and decay. No early release of nuclide is assumed.

Silo. The water flow in the silo is determined by the hydraulic conductivity of the clay barriers and the water flows into the silo through the top. The concrete walls do not limit the water flow. The water in the silo flows mainly through the porous concrete. A fraction of the water flows through fissures in the waste. Nuclides escape from the silo by advection with the water flowing in it and by diffusion through the concrete wall and bentonite around the silo.

Gas is produced in the silo by anaerobic corrosion of metals. When over-pressure in the silo exceeds the pressure needed to open the gas passage in the sand/bentonite barrier, gas can flow out from the silo into the dome and then through fractures in the rock. Stationary conditions are reached, and no water is expelled in this period.

BMA, Water flows into BMA through the cavern walls and the ceiling. Most of the water flow takes place through the space above the concrete construction. Nuclides are transported by diffusion from the waste into the flowing water. A fraction of the water flows through the concrete building, nuclides are then transported out by advection. Products of cellulose degradation may reduce the sorption by forming complexes.

BTF. Water flows into BTF through the ceiling and flows in the unfilled parts of BTF and in fissures in the concrete tanks. Nuclides are transported out by the water flowing through the concrete tanks.

BLA. Nuclides are transported by the water flowing through BLA.

2.1.4 The variation cases during the inland period

Clogging of the evacuation pipes. To expel water from the silo in this period, it is required that gas production starts or increases significantly. Another reason may be an increase of the pressure in the silo caused by the obstruction of an earlier open gas passage. To illustrate the consequences of this situation, it is assumed that clogging of the evacuation pipes takes place. Pressure in the silo is increased until fractures are formed in the evacuation pipes or silo walls. The worst alternative is that the water is expelled through a single fracture at the silo bottom. The sand/bentonite constitutes a hydraulic barrier and sorption on the sand/bentonite retards the transport of radionuclides.

Well direct in a cavern. In another extreme case, a well is drilled directly into a cavern. The probability of this is low since a well in the silo is expected to give little water because of the hydraulic resistance in the silo (Bentonite barrier). Moreover, water in the silo will have a high pH even after 1 000 years. Water from other parts of the repository will not be suitable as potable water due to the large volume of concrete.

2.2 EARLIER MODELLING WORK

Main assumptions used in the calculations:

- Plugs and other actions hydraulically isolate the different parts of the repository and tunnel system. The radionuclide release of each part of the repository is calculated independently.

- Radionuclides are totally dissolved in the water in the packages, except C-14 present as calcium carbonate. The initial radioactive carbon is assumed to be 10 % organic carbon and 90 % inorganic carbon.
- Regarding data for sorption on concrete for the salt-water period, the lowest value between fresh and leached concrete is used. For the inland period, the lowest value between leached and degraded concrete is used.
- Nuclide transport from the repository to the biosphere. No retardation for the salt water period. Retardation is considered for the inland period.
- The direction of the water flow is upward during the salt-water period and downward during the inland period.
- For the inland period, it is assumed that no nuclides have been released during the salt-water period, inventory decreases only by decay.

The main assumptions used in the different parts of the repository for the salt and inland period are listed below.

2.2.1 Silo modelling

The salt water period

- The water flow rate around the silo is 1 l/m^2year. No water flows through the silo.
- The interior of the silo is modelled as a well-stirred tank. Transport resistances in waste matrix, concrete box, and porous concrete are neglected.
- Radionuclide transport takes place by diffusion through the outer silo wall, through the bentonite barrier and into the water flowing in the rock around the silo.
- Sorption on concrete is considered.
- To calculate the diffusion resistance in the outer silo wall, only 0.5 m is considered. It is assumed that 0.3 m is degraded and has a high diffusivity.
- Anaerobic corrosion starts directly after closing. The corrosion rate increases from 0 to 100 % in 20 years.
- The silo is initially water-saturated. About 50 m^3 of water are expelled when gas production start. This contaminated water is retained in the sand layer at the top.
- An over-pressure of 50 kPa is needed to open gas channels in the sand/bentonite. This over-pressure pushes out 140 m^3 of water into the silo wall and bottom.

The inland period

- The groundwater flow is 5 and 15 l/m^2year after 1 000 and 2 500 years respectively.
- The total water flow through the silo is 7 and 25 m^3year after 1 000 and 2 500 years respectively. Water flows into the silo through the top and out through the bottom.
- Sorption on the waste matrix, concrete boxes, porous concrete and interior concrete walls is considered.
- The silo is modelled as a well-stirred tank. Nuclides are transported out with the flowing water and by diffusion through the silo sides and bottom.

2.2.2 BMA modelling

The salt water period

- The total water flow in BMA is 32 m^3/year and is upward. Concrete walls are cracked, therefore they do not resist the water flow.
- The interior of the building is modelled as a well-stirred tank.
- Sorption in the waste matrix and concrete is considered.
- Radionuclides are transported with the water flowing through the concrete building and by diffusion through the concrete walls.

The inland period

- The total water flow in BMA is 320 and 1 000 m^3/year after 1 000 and 2 500 years respectively. The direction of the water flow is horizontal and perpendicular to the cavern. 4 % of the water flows through the concrete building.
- BMA is modelled as a well-stirred tank.
- Sorption in the waste matrix and concrete is considered.
- Nuclides are transport by diffusion into the flowing water in the empty space above the concrete building.
- Calculations were performed with a sorption 50 times lower for those radionuclides that can form complexes with cellulose.

2.2.3 BTF modelling

The salt water period

The calculations are divided into two parts, the first 100 years and the period after 100 years. For the first 100 years:

- The total water flow in BTF is 23 m^3/year and is upward.
- The concrete walls of the tanks avoid water flow through the waste.
- Each tank is modelled as a well-stirred tank. Radionuclides are transported by diffusion through the tank walls. Transport resistance in the concrete around the tanks is neglected.

For the time after 100 years, the concrete hydraulic conductivity has been increased and water flow is possible through the tanks:

- The total water flow is 23 m^3/year and is upward. All the water flows through the concrete tanks.
- Each tank is modelled as a well-stirred tank. Radionuclides are transported by the water flowing through the tanks and by diffusion through the tank walls.

The inland period

- The total water flow is 230 and 800 m^3/year after 1 000 and 2 500 years respectively.
- BTF is modelled as a well-stirred tank. Sorption in the concrete tank and concrete surrounding the tanks is considered.
- Radionuclides are transported by the water flowing through BTF.

2.2.4 BLA modelling

The salt water period

- The total water flow is 23 m^3/year and is upward.
- The whole cavern is modelled as a well-stirred tank. Sorption is not considered.
- Radionuclides are transported by the water flowing through BLA.

The inland period

- The total water flow is 230 and 800 m^3/year after 1 000 and 2 500 years respectively.
- BLA is modelled as a well-stirred tank. Sorption is not considered.
- Radionuclides are transported by the water flowing through BLA.

2.3 PREVIOUS REGULATORY REVIEWS

In 1987, SKB presented to the authorities a Safety Assessment (SSR, 1987). Operation of SFR was allowed with some conditions. SKI and SSI required some improvements in the silo safety assessment (1988). In 1989, SKI and SSI required from SKB an improved assessment for the silo long-term safety. The Deepened Safety Assessment (FSA) was then presented (SFR 91-10). SKI and SSI (SSI 92-07 and SKI 92-16) evaluated the Deepened Safety Assessment for SFR. A short summary of this evaluation is given below.

2.3.1 Criteria used in the review by SKI-SSI

Some criteria used in the evaluation of the SSR and FSA relating to the transport of radionuclides are:
- Barrier functions are described for a sufficiently long time.
- Barrier long-term stability is considered.
- Models used in safety assessments are relevant and sound.
- Uncertainties are considered in the formulation of scenarios and calculated cases.
- The dose for the individual less than 0.1 mSv/year.

2.3.2 Nuclides inventory and waste

Based on the experience of operation, the margins used by SKB for the waste inventory are considered by SKI and SSI to be sufficient. For example, if the waste production from 1987 is extrapolated, the overestimation for Co-60 is by a factor of 3, for Cs-137 by a factor of 5 and for the transuranic elements by a factor of 10. If the 1990 forecast is used, the margins are even larger.

A part of the waste that would be deposited in the silo regarding its origin and activity, will be deposited in BMA (or BTF) to avoid negative effects (gas production and formation of complexes). This increases the activity in BMA and reduces the activity in the silo.

It is important that cellulose is not deposited in the silo. According SKI-SSI, this would be a condition for the operation of the silo. Research into the degradation of cellulose and complex formation is of the utmost importance. Revision of the amount of substances that may generate gas would also be made.

2.3.3 Hydrogeology

According SKI-SSI, in the Deepened Safety Assessment, a discussion of the uncertainties in the given hydraulic properties is missing. SKI-SSI consider that the existence of possible sub-horizontal fracture zones should be investigated. The values of permeabilities used by SKB show a large uncertainty, they could be five and ten times larger for fracture zones and mass rock respectively. However, these permeabilities could be smaller that the values used by SKB. Uncertainty in the measured over-pressures in the Signö zone is also pointed out.

For the inland period, SKI-SSI consider that the water flow may be 4 times larger than the values used by SKB. Additional calculations are required. Uncertainty regarding the water flow direction is also pointed out. Regarding the dilution factor in possible wells, SKI-SSI consider that the dilution factors are large. For example, for a well located directly above SFR, where SKB used a dilution factor of 2, SKI-SSI propose no dilution.

2.3.4 Properties of the technical barriers

Barriers in the repository and their function. According to SKI-SSI, after 500 - 1 000 years, it is possible that the concrete walls will be damaged to such an extent that the concrete walls do not impede water flow. For the silo, vertical water flow could then be established.

Chemical properties. Regarding chemical properties of the concrete, SKI-SSI consider that attack of the concrete by sulphate should be addressed in the perspective of the long- term stability of the concrete. SKI-SSI consider that the uncertainties about concrete mechanical strength are large, but that these uncertainties do not influence the radionuclide release.

Regarding the pH of the water in BMA, SKI-SSI point out that other processes than dilution may contribute to the diminution of the pH.

Regarding the porous concrete, SKI-SSI consider that the Deepened Safety Assessment, FSA, does not show that the property of the porous concrete to carry off the gas from the silo will be retained for long times.

Physical transport properties. SKI-SSI consider that the assumption that the porous concrete will be stable for a long time is uncertain. However, the consequences of a variation in the transport properties of the porous concrete have a small significance.

Mechanical properties of the concrete used in the construction. SKI-SSI point out that the silo will be undamaged for at least 500-1000 years if attack by sulphate is solved in an adequate way. They consider that the value of 280 kPa used as the maximum pressure that the silo may support is too small for fresh and undamaged concrete. For times longer than 1000 years, the silo walls could be damaged and this should be considered when the volume of water pushed out by the gas is determined.

Disturbing processes. SKI-SSI consider that the influence of cellulose in the formation of complexes should be addressed still more. For BMA, the sorption constants for plutonium, americium and technetium are reduced by a factor of 50 to consider the formation of complexes in the presence of a substantial amount of cellulose. SKI-SSI consider that the determination of the factor 50 is not clear and could be improved. The factor for BMA, according to SKI-SSI, should be as large as 75 - 300. For unfavourable cases this factor may be 3 000. Cellulose in the silo and BTF cannot be neglected.

According to SKI-SSI, similar factors could be used for sorption in bentonite and sand/bentonite. For Ni, Co, Fe and Nb no sorption should be used.

2.3.5 Function analysis

According to SKI-SSI, the use of simple models with pessimistic values for the few parameters included in the models may result in too unrealistic a description of the system. They also indicate that the extreme cases accounted for are unrealistic and improbable. Therefore, it is not possible to set a probability measure to the different cases.

According to SKI-SSI, the water flow through the repository is very uncertain. The values used by SKB can be one order of magnitude too large. But, it cannot be ruled out that the values are larger than the values used by SKB.

Each part of the repository is modelled as a well-mixed tank, with maximum release as a consequence. SKI-SSI consider that the release would be significantly reduced if water flows only through certain channels in the waste containers. If the distance between these paths is large, the radionuclide transport is then controlled by diffusion.

Silo. During the inland period, according to SKI-SSI, the flow rate with which water is pushed from the silo will never be larger than the gas production at the then existing pressure. This water flow rate may be compared with the flow rate of the water flowing through the silo. Both the flow rates are of the same order of magnitude, 20-30 m^3/year (water flow rate through the silo proposed by SKI-SSI for the inland period). Therefore, SKI-SSI did not perform any calculation with water expelled from the silo.

SKI-SSI consider 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.

Radionuclides transport by groundwater. According to SKI-SSI, 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. Complexes formed under the high pH in BMA may be destroyed by the lower pH existing outside the repository. These free nuclides may then be sorbed on the fracture surfaces and rock matrix.

Transfer to the biosphere. SKI-SSI point out that a dilution factor of 10 is a careful choice, but that wells nearer than 1000 m from repository are also a possibility. A more pessimistic assumption is to consider no dilution at all, only the dilution inside the repository.

Calculated cases for the inland period. Regarding the inland period, SKI-SSI consider that to get a complete description of the well scenario, a well drilled earlier and nearer to the repository should be studied.

Regarding BMA, SKI-SSI are critical with respect to the water flow through the concrete building. If the whole cavern is modelled as a well-stirred tank, the doses could be one order of magnitude larger.

SKI-SSI performed some calculations to illustrate uncertainties. All repository parts were modelled as a well-stirred tank. Doses were calculated for the case where a well is bored downstream from the repository at 1 500 years.

2.3.6 Further comments in SKI-SSI report

In their evaluation of FSA, SKI-SSI have the following comments

- The most important scenarios were discussed in FSA, but a more complete scenario analysis could have been done.
- The same concept for the land rising is used by SKB in both safety assessments (SSR and FSA). No new description of the course of the land rising has been presented.
- Regarding the variation analysis, no good connection is observed between the scenario analysis and the selection of the calculated cases.
- Alternative models can address uncertainties in the models (mathematical, physical or geometrical). Sensitivity analysis or probabilistic calculations may address parameter uncertainties.
- In FSA (and SSR) two or several extreme cases have not been considered in combination since they are too improbable. SKI-SSI consider that combination of different not too pessimistic cases would be useful. This may be done in a probabilistic way.

In the review, SKI-SSI conclude that some important uncertainties may be reduced if some measures are taken:

- Restriction and control of the amount of organic material into the different parts of the repository.
- Research into the formation of complexes with degradation products of cellulose.
- Regulations for information about the repository.

3 COMMENTS ON PREVIOUS WORK

Some comments are given about the previous Safety Assessments (SSR and FSA) regarding the radionuclide release from the repository and transport to the biosphere.

Hydrology

For the zone where the repository is located, the Darcy velocity is known for the salt-water period, and after 1 000 and 2 500 years for the inland period. For the salt-water period, only one value is assumed. There is no information about the transition from the salt-water period to the inland period (1000 years).

The flow through the different caverns and the silo during the inland period was directly calculated from the Darcy velocity. No hydraulic interactions between the different parts of the repository are considered. The repository is not included in the water-flow modelling.

In the caverns, the waste (or concrete building) occupies a part and the remainder is left empty. No information about the distribution of the water flow between these sections is given.

Formation of complexes

When nuclides form complexes, the solubility is increased and the sorption is reduced. In order to account for it, the value of the sorption coefficient is reduced by a factor of 50. The model on which this assumption is based is not clear. If the model assumes that the nuclide-complex is sorbed with a smaller sorption constant (50 times smaller), this is correct. If the model assumes that a fraction of the nuclide-complex will not be sorbed, the use of the factor 50 is wrong.

Relation between the scenario and the model used

The models used in the release calculations should be based on the scenarios describing the processes that take place in the repository with time. There are some differences (important) between the scenario formulated for a given case and the model used to calculate the release in the same case. Scenario and modelling of BMA is an example.

Long-term behaviour of the barriers

The long-term behaviour of the barriers (concrete, porous concrete, bentonite and sand/bentonite) to a large extent influences the choice of the model to be used. In the models, different criteria are used to consider the variation of the concrete permeability with time.

Conservative models

The models used in both Safety Assessments are too conservative. In a sense, this is an advantage since the model is simplified and some problems may be avoided. In another sense, it is disadvantage, since important processes may be lost.

4 SUGGESTED IMPROVEMENTS

4.1 IMPROVEMENTS

Modelling, to a large extent, depends on the scenarios defined to describe the processes taking place in the repository with time. To improve the modelling in a future safety assessment for SFR, knowledge of several aspects should be increased. Below, some aspects, which it is possible to improve, are discussed in detail. In the following, the term short time will mean the period from the repository sealing to about 1 000 years, the time after 1 000 years will be denoted by long times.

Hydrology

More detailed hydrological calculations are required to improve the modelling at SFR. Large uncertainties regarding the hydrology are found in the earlier safety assessments (SSR and FSA). New groundwater flow calculations including the silo and the caverns should be performed.

For the caverns, the distribution of the water flow between the empty part of a cavern and the concrete building should be determined. This is important for short times, when it is expected that the concrete barriers will remain intact. Calculations similar to those performed for the caverns in SFL should be performed for SFR (Holmén, 1997). For long times, the possibility of using this flow distribution will depend on the long-term stability of the concrete.

Sorption properties of the different barriers

Sorption data for concrete, porous concrete, bentonite, sand, sand/bentonite and concrete used as conditionings for the waste are required. The sorption data should be updated to the "state of the art" values when the safety assessment is performed.

For each nuclide and each material, two values should be supplied: the "best estimate," and the lowest value that may be expected for sorption. For concrete, values for fresh, leached and degraded concrete should be available.

Formation of complexes

Some organic materials (i.e., degradation products of cellulose) may form stable complexes with some radionuclides. The complex formation increases the solubility of the respective radionuclide and decreases the sorption. The former is important if the assumption that the radionuclides are totally dissolved is to be ruled out. For the latter, a consistent and sound approach should be formulated in which the decrease of the sorption constant by complex formation is considered.

Long-term behaviour of the barriers

The barriers may be degraded or damaged with time and water flow through the waste may take place. Moreover, the diffusion resistance of the barriers may decrease if they are degraded. The behaviour of the barriers with time is of vital importance for short times, when sorption and diffusion resistance may significantly retard the release of nuclides with short half-life. If water flow is avoided through the concrete construction, radionuclide transport in the concrete building takes place only by diffusion.

For long times, the function of the repository could be improved if some diffusion resistance could be accounted for. If water flow, for example, occurs through some fractures or preferential pathways through the waste, the transport of radionuclides would be controlled by diffusion.

Gas production

Gas production caused by anaerobic corrosion of metals (i.e. iron) and degradation of organic substances may occur in the repository and a certain volume of contaminated water may be expelled from the repository. The modelling may be improved by a better knowledge of the factors determining the volume and flow rate of the water expelled from the silo. For example, corrosion rates as a function of time, volume of free water in the silo, volume of water easy to remove from porous concrete and concrete.

The silo is designed to allow a given pressure in the interior, 280 kPa. If the silo construction allows for a larger pressure, gas production could expel a larger volume of water if gas passages were clogged (e.g., clogging of evacuation pipes). This should be considered.

Nuclide availability

The concentration of some radionuclides could be controlled by their solubility, due to the limited volume of water in the concrete boxes in the silo. Availability of the radionuclides could also be a factor that controls the concentration.

Nuclide inventory

The radionuclide inventory in the earlier safety assessments is rather conservative. However, to avoid that organic substances are deposited in the silo (complex formation) a part of the waste that would be deposited in the silo was deposited in BMA. This increases the inventory for BMA by a factor of about 2.

4.2 MODELLING ASPECTS

When a system is modelled, an important aspect is the relationship between the uncertainties of the input parameters and the uncertainties in the results from the simulations. This should be considered when data or models are improved. In some cases, the uncertainties in the results are not reduced when the uncertainties of certain "input parameters " are reduced. This is not trivial since the effects of some parameters are very complex and depend on the properties of the radionuclide to be studied.

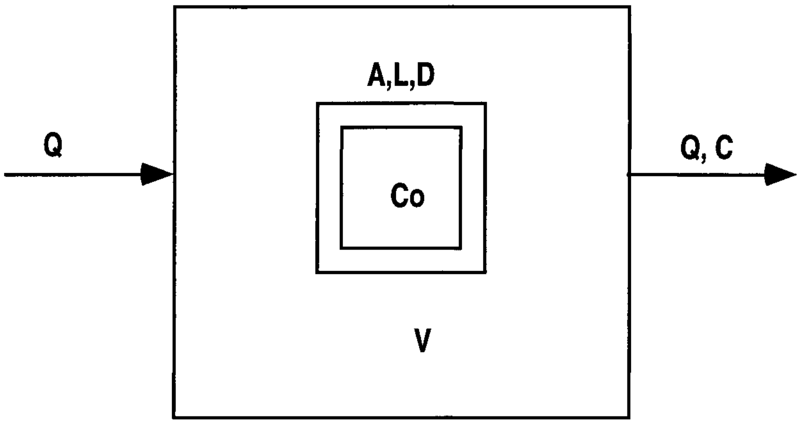
Another difficulty is when radionuclide transport takes place through several pathways in parallel or several steps in series. In the former case, the pathway with the least transport resistance will mainly determine the release. In the latter case, the release will be controlled by the step with the greatest transport resistance. This is valid for radionuclides with half-lives longer than the time constant for the processes taking place in the system. For release of radionuclides with short half-life, sorption plays an important role. This should be considered when the models are improved.

To illustrate this, we can use BMA, where concrete boxes containing the waste are deposited in rooms with thick concrete walls. Water flows in these rooms along the space between the concrete boxes with the waste. In the first case, it is assumed that the diffusion resistance in the concrete boxes is negligible. Radionuclides may emigrate from these rooms by diffusion through the room walls and they may also be transported out by the water flowing in the rooms. If the water flow is sufficiently small, the release will be determined by diffusion. On the other hand if the water flow is sufficiently large, advection will determine the release of radionuclides from BMA.

To continue the discussion, it is assumed that the diffusion resistance in the concrete boxes is not negligible and it is concentrated at the box walls. If this diffusion resistance is sufficiently large, diffusion in concrete boxes will control the release of radionuclides from BMA. Unfortunately in this case, this resistance is small, since the contact area between the concrete boxes and the water in the deposition room is large and the diffusion length is small.

In this description, relative terms (sufficiently small, sufficiently large) have been used. The real values depend on the radionuclides, diffusion coefficients, water flows, diffusion lengths and diffusion surface areas. Once that these values are known, the importance of the different processes should be addressed.

An example is shown as an illustration. Several concrete boxes are deposited in a room at MBA. The total area for the concrete boxes is A m^2 and the diffusion length L m. The diffusion coefficient is D m^2/s and the water flow through the cell is Q m^3/s. The initial concentration in the concrete boxes is C_0, and it is assumed that will remain constant over a long time. The time to equilibrate the walls of the concrete boxes is small. It is also assumed that nuclides escape from the cell only by advection. No diffusion is allowed through the concrete walls of the room.



The differential equation for the outlet concentration is:

$$V\frac{dC}{dt} = \frac{DA}{L}(C_o - C) - QC \tag{1}$$

The solution for C=0 at t=0 is

$$C = \frac{\frac{DA}{L}C_0\left(1 - \exp\left[-\frac{\left(\frac{DA}{L}+Q\right)}{V}\cdot t\right]\right)}{\left[\frac{DA}{L}+Q\right]} \tag{2}$$

From this simple equation, it is possible to evaluate in which case the radionuclide release will be controlled by the water flow rate, Q, and in which case the release will be controlled by diffusion in the concrete walls of the boxes containing the waste.

If the flow rate is small compared with the term DA/L, then the outlet concentration, after a while, will be C_0. This means that the release may be calculated directly as QC_0. This means that advection determines the radionuclide release.

On the other hand, if the water flow is large, the outlet concentration will be $(DAC_0)/LQ$ and the radionuclide release will be DAC_0/L. The release of radionuclide is then independent of the water flow rate and is determined by diffusion. For nuclides with a short half-life, the exponential term should be studied. From this term, the time constant for the process can be obtained.

4.3 MODELLING

A base case should be defined to model the radionuclide release from the repository. This base case should be as realistic as possible for the mechanisms that take place. Simulations using conservative assumptions should be performed. Finally simulations for "unrealistic" extreme cases could be performed.

Models using the well-stirred tank, which is a very conservative assumption, should be avoided for short times.

Sorption is an important mechanism to be considered when the release of radionuclides is modelled. For radionuclides with short half-life, sorption in the barriers retards the transport and the radionuclides may have enough time to decay to negligible activity. Sorption also reduces the concentration of the radionuclides dissolved in the pore water. Sorption should be included in all the materials if suitable sorption data are available.

As pointed out above, the long-term stability of the barriers is crucial in the formulation of the models to be used. When the barriers are quite intact and little or no flow takes place through the containers with the waste, a small release may be expected. In this case, the models should be detailed and include the diffusion resistance in the waste containers. When the barriers are damaged and water flow takes place through the waste containers, no detailed modelling is needed. The well-stirred tank approach could be an alternative, in several cases.

The modelling from the different parts of the repository could be carried out using the program NUCTRAN (Romero et al., 1995a,b). NUCTRAN models the transport of radionuclides from a repository using the concept of compartments. Analytical or semi-analytical relationships are used in locations where other models require a fine discretisation (i.e., equivalent flow rate concept).

4.3.1 Silo modelling

For short times, the release from the silo is small compared with the release from BMA. Improvements in the silo modelling thus have little influence in the total release from the repository for short times. During this period the barriers are quite intact, with the exception of the outer silo wall, which may be damaged by interactions with the bentonite. In the silo, the transport takes place by diffusion. A detailed modelling could be used. The release from the silo into the water flowing in the rock should be calculated using the equivalent flow rate concept.

For long times, the release of radionuclides from the silo is significant compared with the total release from SFR. In this period, the situation is more complicated. The barriers are not intact. The concrete walls have been partially degraded. The concrete is fractured and is not an effective barrier to water flow through the silo. The long-term stability of the concrete determines the model to be used for the silo in this period. The most probable scenario is that water flows in the silo through the porous concrete. This requires that the porous concrete maintains its large hydraulic conductivity with time. Some water flow can, however, takes place through the concrete boxes if they have become damaged. The need for a detailed modelling of the silo interior could be addressed using simple models (Section 4.2). The silo lateral wall and the bentonite around the silo also constitute an effective resistance to the transport of radionuclides by diffusion through them.

4.3.2 BMA modelling

For short times, the long-term stability of the concrete is crucial for the release calculations from BMA. If the concrete is intact, the flow through the concrete building would be small. Most of the flow would take place along the empty space above the concrete building and through the sand-filled space along the concrete building. This should be considered in the model. A detail modelling is important in this period. Retardation by sorption and diffusion resistance are important processes to retard the release of radionuclides with a short half-life.

For long times, the situation depends on the long-term stability of the concrete. If the barriers are damaged, water flow takes place in the concrete building. The most probable scenario is that most of the water flows through the empty space above the building. Part of the flow will take place through the concrete building. For long times, the need for a detailed modelling for BMA should be addressed. At about 1000 years, the radionuclides with a short half-life have decayed to insignificant activity. After that, the diffusion resistance may be important, sorption is in general less important.

4.3.3 BTF modelling

For short times, the stability of the concrete tanks and their special construction should be considered when the radionuclide release is calculated. In other aspects, similar criteria should be applied for BTF as for BMA.

4.3.4 BLA modelling

In BLA, the waste is deposited in steel containers, which will corrode in a short time. Diffusion resistance cannot therefore be accounted for and a well-stirred tank should be used to model BLA. The possible materials for sorption are also scarce. Radionuclides may be sorbed in the concrete used in the floor and on the walls as cover.

4.3.5 Far-Field modelling

The modelling of the transport through the geosphere is influenced by several factors. Some aspects to be considered are:

- <u>Transport paths in the geosphere</u>. According to FSA, in the salt-water period, it is expected that most of the paths from the repository reach the biosphere in the Baltic Sea. During the inland period, the zone above the repository is a zone of recharge. The paths from the repository reach the biosphere, 1-2 km from the repository, possibly in a small lake. With a more detailed hydrology some changes could occur.

- Water flow and flow-wetted surface. For radionuclides that diffuse into the rock matrix and may be sorbed in the rock matrix, the transport in the geosphere is mainly determined by the water flow rate distribution and the flow wetted surface area. For nuclides that are not sorbed in the matrix, flow porosity also influences the transport if the residence time is not sufficiently long. 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 small flow rate and a large flow wetted surface area, retardation by matrix diffusion and sorption may be important.

Migration paths would be modelled in the migration scale hydrogeological analyses. Regional scale and repository scale analyses would be used to judge whether migration paths would 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 salt water 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.

Transport in the geosphere should be modelled for short times, even if the recipient is located directly above the repository.

As discussed by Andersson et al. (1998) there is a general lack of data for the flow-wetted surface and its potential correlation to the 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.

The main migration model used in the SKB deep repository analyses such as SR 97 is FAR31 (Norman and Kjellbert 1990), where the flow field is divided into multiple single flow paths in order to adjust this one-dimensional advection dispersion matrix diffusion model to each migration path. At present, in other SKB project (Alternative models), different model concepts are being considered to study the transport in the geosphere, namely, the Stochastic Continuous Model, the Discrete Fracture Network (code: Fracman-PAWork), and the Channel Network Model (code: CHAN3D). The potential benefits of these different approaches are currently evaluated within the framework of SR 97.

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, Dershowitz et al., 1995; Foxford et al., 1996) and the channel network model (code: CHAN3D, Gylling et al., 1997).

The methodology to be used to model the radionuclide transport in the geosphere would be chosen based in the experiences reached within SR 97.

4.4 DATA REQUIREMENTS

The modelling is, to a large extent, determined by studies carried out in other parts of this project. Once the respective scenarios are defined, the modelling work may start. Due to the uncertainties in the data, various values are required, namely, the "best estimate" to be used in the most probable scenarios and some conservative values.

Some data would be improved before the modelling work, examples are the modelling work are listed below.

Flow-wetted surface. It is a very important entity for the modelling of radionuclides that interact with the rock matrix. The radionuclide transport is locally determined by the relationship flow-wetted surface and water flow rate. For nuclides that are not sorbed in the matrix, flow porosity also influences the transport if the residence time is not sufficiently long. There is a general lack of data for the flow-wetted surface and its potential correlation with the water flow rate.

Sorption data. Sorption data for the materials in the repository are required (Concrete, sand, bentonite, sand/bentonite). For concrete, the sorption data should comprise values for fresh, leached, and degraded concrete. Data for porous concrete and concrete used for waste conditioning are also required.

Formation of complexes. If it is assumed that dissolution is controlled by the solubility, the possible formation of complexes should be considered. Sorption for some nuclides is modified due to complex formation. A conceptual model should be developed to explain the reduction of the sorption by complex formation.

Hydrology. The Darcy velocity in the rock around the silo and the water flow in the caverns are required. For the caverns, the distribution of flow between the empty section and the concrete building should be determined for different ratios of hydraulic conductivities.

Long-term behaviour of the concrete. Concrete properties (Hydraulic conductivity, diffusivity, and even sorption) change with time. The changes of these properties with time should be determined. The same is true of the other barriers in the repository.

5 CONCLUSIONS

The modelling of radionuclide transport from the repository and of the transport through the geosphere depend on several factors, which will be discussed in other parts of the project. Some of these aspects are:

- Long-term behaviour of the barriers
- Sorption and diffusion data.
- Hydrology.
- Formation of complexes.

These and other aspects will decide which specific model is to be used when radionuclide release is to be calculated. For radionuclide transport from the silo and the caverns, NUCTRAN is a suitable tool for these calculations.

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Project SAFE – Prestudy

Appendix A6:

Biosphere

Ulrik Kautsky

SKB

Ulla Bergström

ECOSAFE

Contents

1 INTRODUCTION

This report addresses the biosphere of the safety analysis for SFR. It gives a brief description of earlier performed studies, summarises the review from the authorities and gives suggestions for improvements of the methodology for a new analysis.

In previous safety assessments the biosphere had a minor role or it was described in a stylised way. However, the coming regulations require that the effects should be described for a realistic biosphere as well as potential effects on other organisms than man. This requires a more realistic model of the biosphere based on ecological know-how rather than isolated simplified food-pathways. The current safety assessment for SFR concerns a real site where the different components of the ecosystem are obvious for laymen. Thus presented models must be clearly related to the surrounding natural system to receive common acceptance. Moreover there is a unique opportunity to gain new knowledge from an existing system which can be used for the safety assessment of a high-level repository. Although it will be necessary to develop new methods and collect new field-data the main work can been done with existing knowledge among ecologists and geologists and data gathered in the area for other purposes.

Moreover in the safety analysis of the repository of high level waste (SR97) new methods have been developed which will be used for the SAFE assessment.

2 METHODS USED IN PREVIOUS SAFETY ASSESSMENT OF SFR-1

The biosphere was handled in the first safety analysis, 1987 (SKB 1987). It was based on Bergström et al, (1987) where model descriptions and results were given. The results from FSAR (SKB 1987) and the review from the authorities lead to a study (Hesböl et al, 1991) dealing explicitly with the problems of C-14. In the revised FSAR (SKB 1993) the results from Bergström et al, (1987) were used with an extension of the well scenario.

2.1 SAFETY ASSESSMENTS, BASE SCENARIOS

Two main scenarios were handled - one for leakage of the nuclides directly to the coast (brackish or marine scenario) and one where the releases occur to an inland area after land-rise. The biosphere was modelled by compartment theory, where the biosphere components were divided into physical areas with uniform properties. The exchanges of nuclides between those compartments were 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 global population from calculated releases from the repository. Therefor the model was divided into four spatial scales to be able to simulate the turnover of the radionuclides from the local to the global area. The four scales were:
- Local zone
- Regional zone
- Intermediate zone
- Global zone

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

In **The regional zone,** the dispersion of radio-nuclides in an expanded area around the point of discharge was considered. Recipients were the entire Öregrundsgrepen and the lake for the coast and inland scenario, respectively. 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.

Some additionally analyses were performed to the calculation of doses to critical group and population

An uncertainty analysis was performed addressing Cs-137 for the coast scenario. Several parameters were varied and analysed with PRISM (Gardner at al, 1983). The analysis showed that human consumption and bioaccumulation were the main contributors to the uncertainties in calculated dose.

Variation analysis was performed for a well scenario. This showed that dilution volumes gave the major contribution to the variations in dose estimates. Finally, doses for Pu-239 and Pu-240 were estimated, assuming that all released activity was trapped in lake sediments. These sediments were then used for agricultural purposes. The concentrations of the elements in this soil were lower than those obtained from irrigation of soil with water from the well.

2.2 MODEL DESCRIPTION

2.2.1 Model structure

Local zone

The main recipients Öregrundsgrepen and the lake were subdivided into compartments for water and two sediment layers (Figure 1). Between these compartments exchange of nuclides due to water turnover and particle fluxes were considered.

Homogenous mixing was assumed within the compartments, which made it necessary to subdivide the sediment in two compartments. One compartment was the upper sediment (0-10 cm), where oxidising conditions prevail and where an exchange with the overlying water occurs. The other compartment was the deeper sediment with reducing conditions, low bio-turbation and low water-exchange.

The surrounding agricultural area was divided in three compartments, where the upper layer (ca 0-30 cm deep) consisted of the root zone where agricultural practices take place. The next reservoir was the soil beneath the upper surface, and the third level represented the ground water (Figure1).

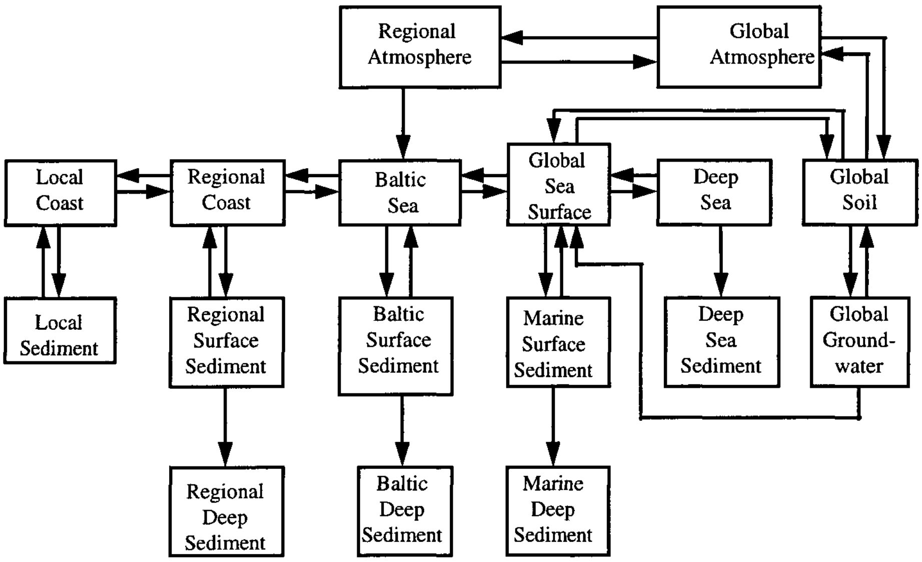

Figure
The structure of the BIOPATHs compartment model for the coast scenario

Figure. 1 The structure of the model for the coast scenario.

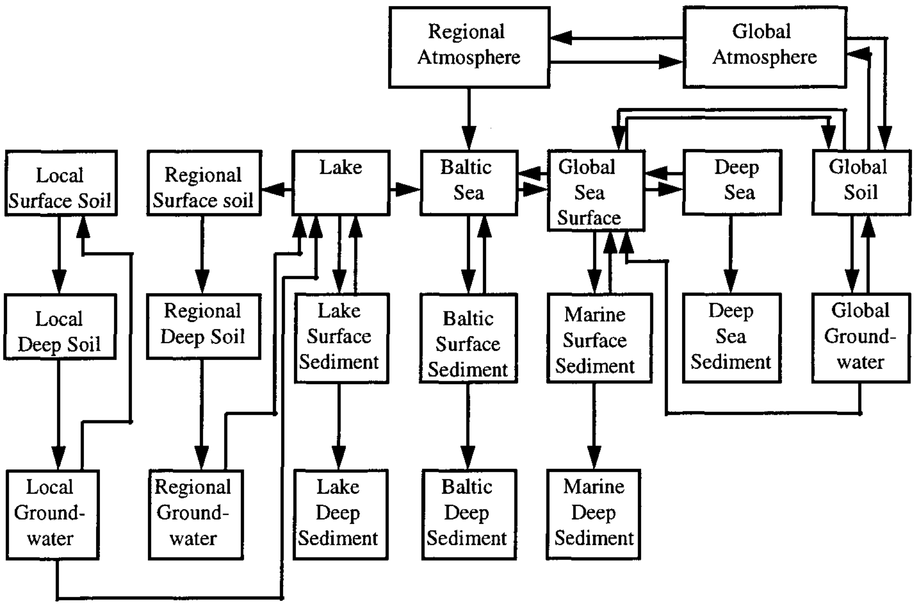

Figure
The structure of the BIOPATHs compartment model for the inland scenario

Figure. 2 The structure of the model for the inland scenario.

Intermediate zone

The intermediate zone was not dependant on any primary recipient and consisted of the Baltic Sea and its sediment (Figures 1 and 2).

Global zone

The global zone consisted of five compartments. Two compartments for the water in the oceans, one compartment represented the upper, well mixed, surface water down to approximately 150 m, and the other compartment represented the deeper parts of the oceans (Figures 1 and 2).

The other compartments were the ocean sediments, global groundwater and global terrestrial soils. The compartment systems applied are shown in Figures 1 and 2 for the coast and inland scenario, respectively

2.2.2 Model function

The exchange of radionuclides between the compartments was described by nuclide-specific transfer coefficients. This lead to several first order linear differential equations which were solved numerically by the program BIOPATH (Bergström et al, 1982). The processes included in the models for the turnover of nuclides were as follows

- Turnover of water in all water compartments
- Transfer to sediments from water
- Leakage from upper soil to deeper soil
- Leakage from deeper soil to groundwater
- Outflow of groundwater to surface water
- Irrigation of farming land
- Resuspension from sediment to water
- Resuspension from soil (global area)
- Deposition from regional and global atmospheres

The element specific rate constants were obtained from water flows and a retardation factor for the soil compartments. The retardation was described with K_d-factors in combination with porosity and densities of the soils. The transfers to the sediments were obtained by area-specific mass sedimentation rates coupled to K_d-values, concentration of suspended matter and actual mean depths. Resuspension from the sediments back to water and transfer to deeper sediments were generic values in common for all elements.

2.2.3 Exposure pathways to individuals and population

In the assessments only internal exposure pathways were considered. This simplification was justified from the experiences of earlier performed calculations of exposure from disposal of high-level waste (Bergman et al, 1979, Bergström 1983). The internal exposure pathways were those connected to consumption of water and agricultural products or consumption of marine products. The exposures were calculated in a traditional way (IAEA, 1982), that is use of constant parameter values for root-uptake and transfer to milk and meat, respectively. They were also briefly commented and references were given to the sources.

Radioactive nuclides, which may occur in food, are transported in the food chains via a varying number of paths.

- Crops take up radionuclides from the soil by their root system and may be contaminated through deposition on the vegetation surface.
- Occurrence of radionuclides in meat and milk is due to the animals' metabolism from intake of contaminated cattle food, soil and drinking water.
- Fish and other marine products absorb radionuclides from surrounding water and through food.

Crops

The calculated uptake of radioactive nuclides in groceries was based on concentrations of radionuclides in soil and water as a function of time. It was supposed that a steady-state exchange between vegetation/soil and fish/water prevails. Contamination of vegetation surfaces due to irrigation or deposition of resuspended material carrying radionuclides was also included.

Cattle

Animals may take in radioactivity through their food as well as by consumption of water and soil while grazing. The ingested nuclides are then transferred to milk and meat. That transport was described by distribution factors (Ng et al, 1977, 1979), assuming steady-state conditions. These factors expressed as day per litre or kg give concentration of respective element in milk and meat from a daily amount of intake.

Well

In the well scenario, radionuclides reached man through drinking water both directly and via cattle. The wells were assumed to have such a low water capacity that they were not used for irrigation of agricultural crops and thus only irrigation of green and root vegetables was taken into account. However, it was assumed that the critical group was also exposed to contaminated fish from the lake. In addition lake water was used for irrigation of adjacent farming-land.

Lake

When the lake was recipient, it was assumed that the water in the lake was utilised as drinking water for both humans and animals. Furthermore, the assumption was that the lake water was used for irrigation of agricultural areas. This means that man was exposed from all foodstuffs, all of which was locally produced.

Final assessment

In the final assessment the exposure to critical group occurred simultaneously from the well and lake water in the inland scenario. The exposure pathways to critical groups considered for each case are summarised in Table 1 where the compartments used for calculation of each exposure pathway are shown.

Table 1

Path of exposure	Inland scenario	Coastal scenario
Inhalation	Regional surface soil	
Drinking water	Well	
Milk	Well, soil	
Meat	Well, soil	
Green vegetables	Local surface soil	
Cereals	Regional surface soil	
Root vegetables	Local surface soil	
Eggs	Well, regional surface soil	
Fish	Lake	Coast

Exposure pathways for the **regional zone** were considered in a similar way as for the local zone. There was however no exposure of the regional population through the part of Öregrundsgrepen which was primary recipient in the coast case, or from the well in the inland case as doses to critical group were not included in the calculations of collective doses (Bergström et al, 1987). Potential values of annual yield for the agricultural area and Öregrundsgrepen were therefore utilised to calculate doses.

The exposure pathways from the **intermediate zone**, i.e. the Baltic Sea, were through consumption of fish and algae products. Algae were considered because they may become an important protein source in the future.

In the **global zone**, all terrestrial exposure pathways were taken into account, as well as the same marine exposure pathways as for the Baltic Sea.

2.3 REGULATORY REVIEW

SSI (Swedish National Radiation Protection) performed a review of the FSAR 1988 (Bergman C et al, 1988). They 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 underestimated the collective dose for C-14 in comparison to the calculations performed by SSI (Bergman C et al, 1988). This error in combination with the high importance of exposure from C-14 lead to that a specific study of C-14 was initiated. The authorities pointed also out that realistic estimates of doses should be made.

After the in-depth analysis 1991, SSI and SKI made a joint report reviewing the safety analysis (SSI, 1992). The main comment from that review pointed out that documentation and model descriptions should be improved as well as reasons for uncertainties should be discussed. However, the authorities focused considerably on location of wells, as wells were deemed to give the highest exposure.

2.4 COMMENTS ON PREVIOUS WORK

Since this assessment was performed much more interest has been devoted to radio-ecological models and their reliability. International studies have been carried out to study such questions (BIOMOVS 1, BIOMOVS 2, VAMP). Reasons for uncertainties and mistakes in results from models for environmental assessments may emanate from the following main reasons (IAEA, 1989).

- specification of the problem (scenario)
- formulation of the conceptual model
- formulation of the computational model
- estimation of parameter values
- calculation and documentation of results

Specification of the problem

The main target for the assessments were estimations of the doses to man. Nowadays much more interest is coupled to the consequences to biota than earlier. A new assessment should not only consider man but also estimate the consequences to flora and fauna according to criteria from the authorities.

The consequences to man were addressed by applying one coast and one inland scenario. The nuclides entered the biosphere however directly to the coastal water or into the lake water. Processes in the geosphere/biosphere interface were not taken into account, mainly because of lack of knowledge of processes at this interphase where chemical conditions may drastically change causing strong gradients. The neglecting of this leads to a direct mixing of the elements transported with groundwater in considerable amount of water. Lake and coastal water may, however, not be the only entrance points of groundwater to consider others are running waters, wetlands and soils.

During the transport with groundwater nuclides may also reach soil and thereby be able to give rise to exposure of man. This area though earlier neglected was subject for one scenario within the BIOMOVS 2 study. In a lysimeter experiment some radionuclides were introduced into the ground water table that was kept at a constant level (BIOMOVSIIc, 1996). These radionuclides were detectable along the whole soil profile due, mostly, to capillary rise and biological processes.

Formulation of the conceptual model

The conceptual models were generic although the site is a well-investigated area. In order to improve the assessments and give more reliable results more site-specific information should be used when designing the model and select input parameter values. More emphasis is needed for developing of the conceptual model and the documentation behind it. More considerations need to e.g. be taken about biotic processes also for the turnover of nuclides in the biosphere.

It is evident that the C-14 model and parameter values need a review. The model designed in Hesböl et al, (1991) considered a time-dependent uptake of C-14 at steady-state to fish. The uptake was related to the concentration of C-14 in water. This model was evaluated against experimental data from a lake in Canada where C-14 had been added (BIOMOVS IIb, 1996). The first comparison performed within BIOMOVSII showed that the model results were lower than the observed values. As C-14 is one dose-dominant nuclide it is important that calculated values of levels in fish are not underestimated

In the coast scenario the applied model is very sensitive to the water turnover. The water turnover in Öregrundsgrepen was calculated with a hydrodynamic model which generated turbulence and velocity fields to a dispersion model which calculated the dispersion of soluble and particulate matters (Rahm et al, 1989). The results were compared to measurements of currents in the area. The comparison showed a reasonable agreement between the model simulations and the observations. A comparison was also carried out with the water turnover rates applied in a box model for the area (Sundblad et al, 1983). The conclusion was that the water turnover rates in the box-model were in agreement with the ones obtained with the sophisticated methods. But they pointed out that the model approach did not take uptake into biota into account. Other limitations were the neglection of diffusion of elements in the sediment/water interface.

Formulation of the computational model

The numerical approach used has been verified in national as well as international studies, showing that this step does not cause any substantial contribution to uncertainties (Persson et al, 1975 and Klos et al, 1993).

Estimation of parameter values

In the study (Bergström et al, 1983) some estimates were made for Cs-137 for the propagation of varying parameter values in the model calculations. Such analyses should with preference be used in all assessments, especially as the literature in general shows wide ranges for many parameters, e.g. K_d (Sheppard et al, 1990). By applying such techniques it is also possible to identify the major components to the uncertainties in order to make improvements for these critical parts of models.

2.5 SUGGESTED IMPROVEMENTS

- The assessments should be based on more site-specific information. The area is well studied since the construction of the nuclear power plants.
- A better description of morphology 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.
- Studies of local accumulation zones.
- Evaluate in more detail the turnover of C-14 in aquatic systems.
- A new conceptual modelling could be carried out taking ecological interactions into account in a more detailed way
- All the assessments should be given with estimates of the uncertainties in results
- Review of data and updating of the dose-conversion factors
- Consider the exposure to flora and fauna from released radio-nuclides

3 METHODS PROPOSED FOR THE BIOSPHERE IN PROJECT SAFE

The previous model used in the safety assessment is described earlier. Since the last safety assessment the methods have developed, and the aim is to apply at least the same methods for the biosphere in this assessment as in SR97. That means that a modularised concept is introduced. However, there is a need of further improvement of the methods as outlined in the introduction. Moreover is the concept with concentration factors currently used in the biosphere modelling questionable due to high variations and uncertainties 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 a better knowledge of the biological structure of the system is necessary. Moreover the transport of radionuclides in the loose deposits needs more attention. This includes modelling of flow of water as well as the flow of deposits. The loose deposits were estimated in Sigurdsson (1987), but the information was not used.

In the time-perspective of the SAFE assessment we need also to consider feedback-mechanisms from the biosphere to the geosphere, affecting flow-paths and geochemistry.

3.1 IMPLEMENTATION OF THE SR97 METHODS

The current biosphere models, dose-factors and interaction matrix developed and used in the current safety assessment of the high-level waste (SR97) needs to be revised for SAFE. The changes and development of these are described in Section 2. Moreover the knowledge developed in the projects described below will be incorporated in SAFE.

3.2 THE STRUCTURE AND FLOW OF LOOSE DEPOSITS

A detailed geological field survey of the sea-floor above SFR was performed (Sigurdsson, 1987). This detailed information about sediment depth and particle size has not been used in further estimates, however it gives important information on residence time, sorption and transport pathways of elements. Moreover this detailed information is useful to estimate particle transport due to resuspension. The topography and hydraulic conductivity of the material are also important parameters for the hydro-geological model. The sediment type is also of importance to determine potential dominant ecosystems.

The local area above SFR is well described regarding sediments (Sigurdsson 1987), however to which extent is this useful for estimates over a larger regional area (i.e. Öregrundsgrepen)? This needs to be evaluated primarily by studies of available data from sea-charts, core samplers and other sources. If this is not sufficient some echo sounding transects are necessary to show rock bottom vs. sediment bottom.

Due to land uplift the loose deposits will be subjected to erosion and resuspension by waves, ice and wind. These processes are active on deposits down to several tenths of meters below sea level. The resuspension of particles has several important consequences: 1) radionuclides accumulated in the sediment several hundreds of years are rapidly released and causing doses orders of magnitudes higher; 2) accumulated radionuclides from several discharge-points are resuspended and concentrated in the deepest basins and thus causing higher doses directly due to the concentration or in a later stage when the former sea-floor is used for farming; 3) The topography above land will not be the same as the sea-floor topography which will have effects on the pressure gradients affecting the hydrogeological conditions.

This means that these processes need further studies and that it is difficult directly to use the below sea topography as a model for future land topography. Moreover during the land uplift process the water turnover will change, because land and shallow parts will retard water movement.

This needs a development of a model calculating the resuspension of sediments dependent primarily on wave effects, which is estimated from predominant winds and fetch. Probably fetch (i.e. length of open water in each wind-sector) will be the most important parameter. The fetch will vary dependent on how much land will be elevated in the surrounding area. An estimate will be allowed when taking the shore-level displacement in combination with topography into account. Ice and wind strength are dependent on climatic conditions, which are difficult to predict, and thus it is suggested that the present day situation is used with some possible extreme variations.

The results will be presented with maps showing the topography and bays, lakes and accumulation areas at different stages during the coming 10 000 years. The model can be calibrated with the recent deposits on land in nearby area and a reconstruction of sea level changes and fetch.

Thus the most important information is the change of the topography, the accumulation alternative dispersion of particles and the resulting substrate when the sea-floor is converted to land.

3.3 WATER-TURNOVER AND OTHER TRANSPORT

The water-turnover was identified as important factor for the dose-estimates. The water-turnover also affects the transport and dispersion of radionuclides in solution and bound to particles (see above) and organisms. A new method to calculate the turnover will be used in the area (Engqvist, 1993, 1997). The turnover will also be recalculated at different stages during the land-rise. The effects of migrating organism and food stuff will be incorporated in the transport calculations.

3.4 THE STRUCTURE AND SUCCESSION OF THE ECOSYSTEM

There is no site specific description of the ecosystems surrounding the SFR. However , some work has been done in the surrounding area on land and in the water, e.g. fish stocks by Kustvattenlaboratoriet, the sediment ecosystem from monitoring programs (PMK), and projects in the biotestlake at Forsmark. This information must be compiled to be useful to describe the structure of the ecosystem. Especially there is a need to describe the aquatic system and this probably requires a field survey to quantify and describe the dominant components of the ecosystem in the area above SFR as well a comparison with Öregrundsgrepen. Probably the most important communities will be filter-feeders, deposit-feeders and macrophytes and fish (se below). Moreover the total carbon content in sediment and water is important to estimate both for the carbon flow as well as for dilution effects of ^{14}C. Some of this general information is available from the monitoring programs around Forsmark nuclear power station and especially from the work that has been done in the biotest basin. The calculation of fetch and sediment-type described above will be useful for generalisations of ecosystem structure.

After the ecosystem structure of today is described, the succession to future states must be predicted. Land-uplift is the most dramatic change which is possible to predict (Påsse, 1996). Especially the change from sea to inland, alternatively lake is important. But changes in the salinity of the sea, acidification of lakes and land-use are more difficult to predict. To estimate the effects of these processes some extreme variations needs to be tested, e.g. a marine and a freshwater- area. Moreover the cultural habits of man will be of importance, where the most significant variable is the size of the life support area. A highly self sustainable society has a small life support area, i.e. low dispersion and mixing of radionuclides, whereas a society as today has a large life support area, i.e. high dispersion and mixing of radionuclides through food supply from different regions. Here again some examples of extreme variation of the life support area can be considered.

3.5 EFFECTS ON FAUNA AND FLORA

The coming regulations from SSI stress that effects on other organisms than man should be estimated. From the ecosystem model described below, compartments will be identified which have high mass-flow and storages 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, which are occurring in the area and are well investigated concerning organic xenobiotic substances (e.g. PCB, DDT).

3.6 THE ECOSYSTEM MODEL

The aquatic part of the ecosystem model is shown in Figure 3. It is subdivided in a physical and a biological submodel. The physical model (Figure 3a) describes sedimentation, resuspension and water-exchange mainly affected by physical and chemical processes. The radio-nuclides 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 radio-nuclides. Probably a large part of the radio-nuclides will be stored there until the sediment is resuspended. The resuspension is affected by wind and sea-level changes. These two factors have large effects also on the water turnover. The sinks of radio-nuclides are radioactive decay, export with the water or suspended particles and through release to the atmosphere (e.g. ^{14}C).

The ecological part of the model (Figure 3b) is more complex, however several of the compartments (the boxes) can be omitted or aggregated when the structure of system is described as outlined above. The main uptake pathways is through consumption of food contaminated with radio-nuclides. Filter-feeders (e.g. bluemussels, Baltic Sea mussel) receive most of the substances through feeding of suspended particles (plankton and detritus particles) and from gill uptake of dissolved radio-nuclides. Filter-feeders can filter the water in an entire bay within 1-2 weeks in most coastal areas (Kautsky 1995). Filter-feeders are common food source for fish and birds. Moreover filter-feeders deposit large amounts of faecal pellets which means that filter-feeders are an important gateway to transport suspended or dissolved radioactive substances to the bottoms. 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 radio-nuclides.

Another potential important transport mechanism it through benthic plants (e.g. algae and reeds). Radio-nuclides sorb to the plant or are actively assimilated. The plants then detach or fragment and sink to deeper bottoms, where they are decomposed and consumed by organisms, which then are consumed by fish. This is an important pathway in most coastal areas. A minor flow of matter is through grazers feeding on plants, however, this can lead to high accumulation in grazers which then are consumed by fish. Plankton are also efficient transporters of radio-nuclides transferring substances to larger plankton and then to fish or to filter-feeders .

The benefits with ecosystem models are that the transfer of radio-nuclides 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. Moreover also 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.

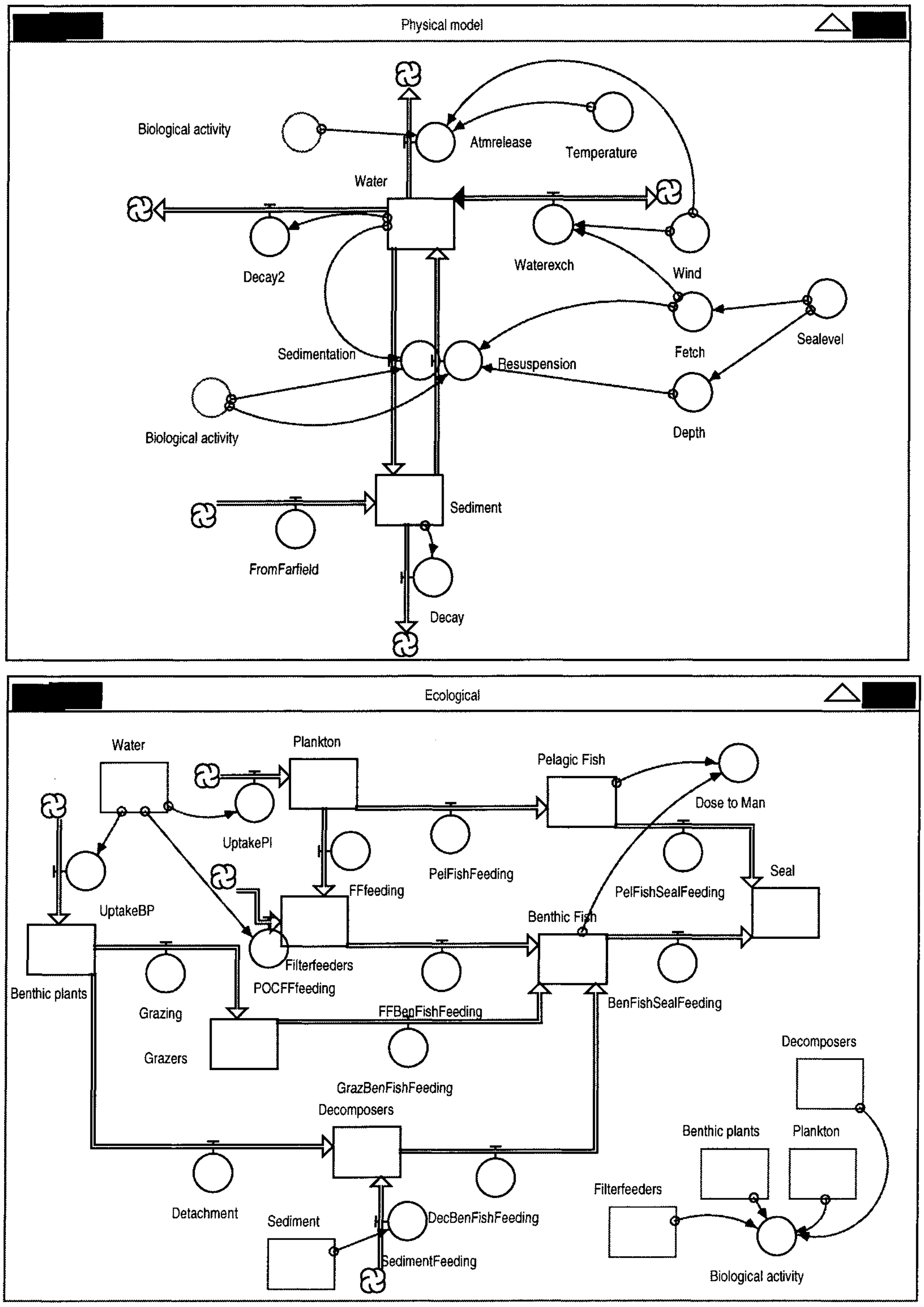

Figure 3 Conceptual ecosystem model of an aquatic ecosystem. Upper part shows physical processes, lower part biological processes. Boxes are reservoirs, thick arrows fluxes of matter, thin arrows relationships to parameters and external variables (circles).

3.7 INTERACTIONS GEOSPHERE - BIOSPHERE

As mentioned earlier at the interface between the geosphere and biosphere important processes occur which have been not addressed in earlier safety assessments. Moreover the transport of nuclides in the loose deposits, i.e. near surface hydro-geology, needs further consideration. 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 solved with an integrated approach between geosphere and biosphere.

4 SAFETY ANALYSIS

The ecosystem model will predict where radionuclides will accumulate. This enables that doses 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 gives information on transport of particles and input-data to the far-field model. This affects the dispersion and transport of radio-nuclides together with the water-turnover. The interdependence of the project is sketched in Figure 4.

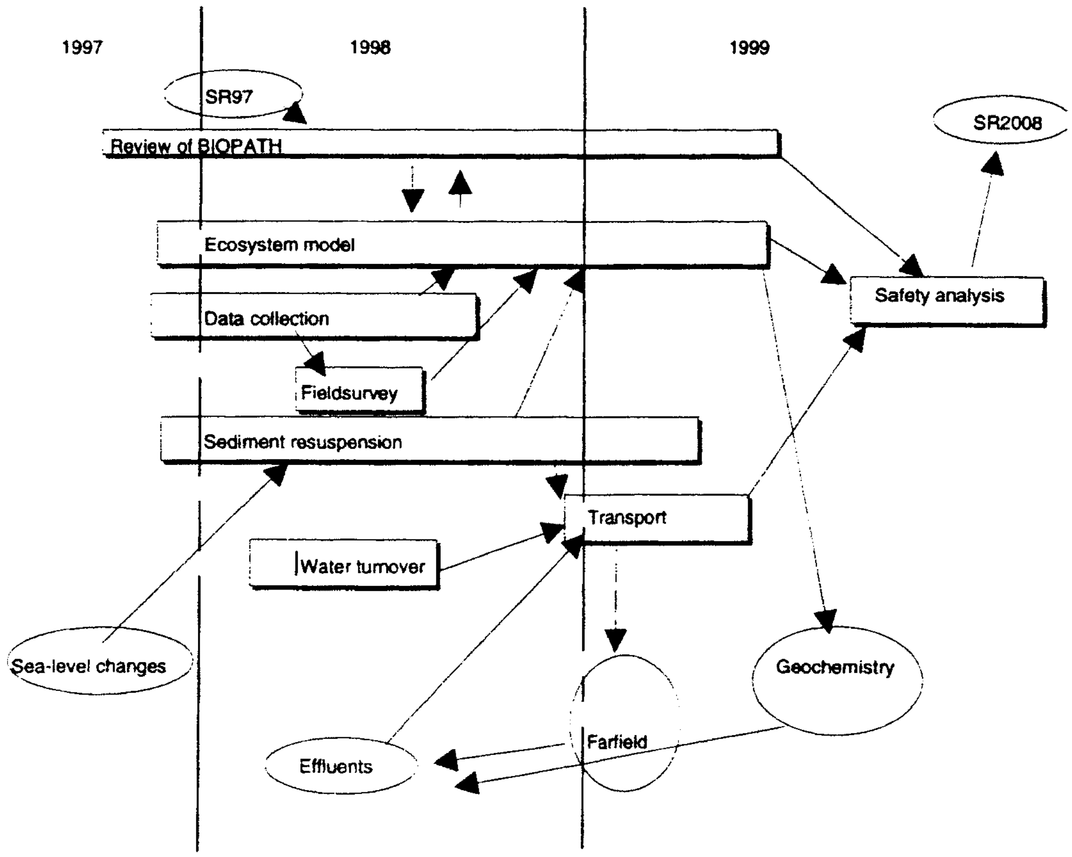

Figure 4 Relationships between subprojects in the biosphere. Rectangles show subprojects within the biosphere program, ellipses external data.

5 SUMMARY

There has been a considerable development of models used for describing the turnover of radionuclides or other pollutants in the biosphere. New regulations require realistic assessments and description of effects on fauna and flora. Thus the use of trophic transfer models will be a more appropriate way to model the biosphere. These models take all accumulations of radionuclides in the ecosystem into account, not only direct pathways to man. Thus these model must be developed for this area. Moreover the turnover of loose deposits needs to be modelled. To able to use these models there is a need to collect data on sediment composition, ecosystem structure and potential changes due e.g. sea-level fluctuations. These data will be collected from literature and where it is necessary complemented with field surveys. In some cases new models need to be developed. The integration of the geosphere and biosphere models is identified as an important issue.

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