Main Report

Summary

Deep repository for spent nuclear fuel

SR 97 – Post-closure safety

November 1999

Foreword

During the past three years, SKB has carried out an assessment of the long-term safety of a deep repository for spent nuclear fuel. The results of the project are reported in Swedish as "Djupförvar för använt kärnbränsle; SR 97 – Säkerheten efter förslutning". This report is an English translation titled "Deep repository for spent nuclear fuel; SR 97 – Post-closure safety". The Main Report in its complete form consists of two parts with accounts of premises, methodology, analyses, results and conclusions. In addition there is a detailed summary which contains, among other things, the entire conclusion chapter from the complete version.

The report is primarily written for experts, but parts of the text should be of interest to non-specialists as well.

Allan Hedin has been responsible for methodology and for coordination of the different parts of the project, has written the summary, and has acted as writing editor for the complete main report. Patrik Sellin has dealt with near-field subjects. Anders Ström and Jan-Olof Selroos have been in charge of geosphere-related matters, and Ulrik Kautsky has been responsible for the biosphere. Lena Morén has worked with the climate and intrusion scenarios, while Fredrik Lindström has carried out the radionuclide transport calculations.

Many other individuals inside and outside SKB have also contributed in various ways to the project. If any are to be given special mention, the difficult choice falls on Johan Andersson of Golder Grundteknik, who participated as an expert in both geosphere matters and safety assessment in general, and Harald Hökmark of Clay Technology, who has worked with mechanical questions in the geosphere.

SKB is responsible for all judgements and conclusions in the report.

Stockholm, November 1999

Tönis Papp
Director Safety and Science

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Summary

Purpose and premises

In preparation for coming site investigations for siting of a deep repository for spent nuclear fuel, the Swedish Government and nuclear regulatory authorities have requested an assessment of the repository's long-term safety with the following purpose: "…to demonstrate that the KBS-3 method has good prospects of being able to meet the safety and radiation protection requirements which SKI and SSI have specified in recent years."

SR 97 is the requested safety assessment. The purpose is to demonstrate by means of a systematically conducted analysis whether the risk of harmful effects in individuals in the vicinity of the repository complies with the acceptance criterion formulated by the Swedish regulatory authorities, i.e. that the risk may not exceed 10^{-6} per year. Geological data are taken from three sites in Sweden to shed light on different conditions in Swedish granitic bedrock. The repository is of the KBS-3 type, where the fuel is placed in isolating copper canisters with a high-strength cast iron insert. The canisters are surrounded by bentonite clay in individual deposition holes at a depth of 500 m in granitic bedrock.

The assessment applies to a closed repository for spent nuclear fuel and thus does not include either safety during operation or safety of the repository for long-lived low- and intermediate-level waste. These matters are dealt with in separate reports.

Methodology

The methodology in the assessment entails first describing the appearance of the repository when it has just been closed and then analyze how the system changes with time as a result of both internal processes in the repository and external forces. The future evolution of the repository system is analyzed in the form of five scenarios. The first is a base scenario where the repository is postulated to be built entirely according to specifications and where present-day conditions in the surroundings, including climate, are postulated to persist. The four other scenarios show how the evolution of the repository differs from that in the base scenario if the repository contains a few initially defective canisters, in the event of climate change, in the event of earthquakes, and in the event of future inadvertent human intrusion. Repository evolution is broken down into thermal, hydraulic, mechanical and chemical processes, and the ultimate purpose of the analyses is to evaluate the repository's capacity to isolate the waste in the canisters, and to retard any releases of radionuclides if canisters are damaged. The time horizon for the analyses is at most one million years, in accordance with preliminary regulations.

Base scenario

By means of model studies and calculations, the base scenario analyzes how the radiotoxicity of the fuel declines with time, the repository's thermal evolution as a result of the decay heat in the fuel, the hydraulic evolution in buffer and backfill when they become saturated with water, and the long-term groundwater flow in the geosphere on the three sites.

Mechanical stresses on the canister stemming from groundwater pressure and swelling pressure from the buffer are examined, along with the long-term mechanical stability of the geosphere. The chemical evolution of bedrock and buffer, as well as corrosion of the copper canister, are also analyzed.

The overall conclusion of the analyses in the base scenario is that the copper canister's isolating capacity is not threatened by either the mechanical or chemical stresses to which it is subjected. The safety margins are great even in a million-year perspective.

Canister defect scenario

The internal evolution in initially defective canisters and the possible resultant migration of radionuclides in buffer, geosphere and biosphere are analyzed in the canister defect scenario. The result is estimates of dose and risk that can be compared with the acceptance criterion for a deep repository.

The scenario first shows that criticality cannot be expected to occur in the repository.

Analyses of the hydromechanical evolution in a damaged canister when water enters show that even the damaged canister prevents the release of radionuclides for a very long time, since intruding water is consumed by corrosion of the cast iron insert.

Dissolution of the fuel and solubility conditions for radionuclides released from the fuel are studied in analyses of the chemical evolution in a damaged canister. Model calculations show that hydrogen gas generated by corrosion of the cast iron insert contributes towards keeping the rate of fuel dissolution low.

Groundwater flow is studied on a local scale on the three sites. The analyses show that variation in results stemming from the natural variability in the rock often overshadows the variation caused by both differences between model concepts and uncertainties in boundary conditions, fracture structure, etc.

Radionuclide flux in the biosphere is modelled for a number of ecosystems, e.g. well and peatland. Peatland gives relatively high doses as a consequence of accumulation of e.g. Ra-226.

Data from the above-mentioned studies are then used for calculations of radionuclide transport in canister, buffer, backfill and geosphere. Releases from the geosphere are converted to doses in different ecosystems. Both reasonable and pessimistic values are estimated for all input data to the calculations, and in a few cases statistical distributions as well.

With reasonable data, the doses on all sites lie far below the dose limits that can be derived from the official acceptance criteria. The influence of uncertainties in data is analyzed by systematically substituting reasonable data for pessimistic data and studying the calculation result. The variation in flow-related data in the geosphere has the greatest impact on the result, followed by data uncertainties for the biosphere. Other conclusions are that our understanding of fuel dissolution needs to be improved, and that the probability and size of initial canister defects that escape quality-control inspection is difficult to estimate.

In order to obtain a risk measure that can be directly compared with the acceptance criterion, risk analyses in the form of simplified probabilistic calculations are also performed. The risk analyses show that all sites lie well below the acceptance criterion.

The maximum risk for release to a well is never more than 0.5 percent of the acceptance criterion, even when the calculations are extended a million years into the future. The same applies to releases to peatland for times up to 100,000 years, while the maximum risk here grows to about one-tenth of the acceptance criterion at the least favourable site at times after 100,000 years.

Climate scenario

The consequences of future climate change are explored in the climate scenario. Today's climate is relatively warm by historical standards, and future changes are expected for the most part to lead to a colder climate as a consequence of cyclical variations in insolation. A conceivable sequence of events, including severe glaciation, on each of the three sites is sketched for the coming 150,000 years.

The repository system's thermal, hydraulic, mechanical and chemical evolution under the changed conditions in the surroundings is studied in the form of a comparison with the evolution in the base scenario.

In the climate scenario as well, the overall conclusion is that the isolating capacity of the copper canister is not threatened by either mechanical or chemical stresses. The mechanical stresses are larger than in the base scenario, mainly due to higher rock and groundwater pressures in connection with a glaciation. The chemical stresses are roughly the same, partly because oxygen-containing groundwater is not expected to reach the canister. The strength calculations for the canister may need to be refined with more realistic, inhomogeneous material properties, and buffer erosion with extremely ion-poor groundwater compositions may require further study.

As far as the retarding capacity of the repository is concerned, for example in the event of initial canister damage, the most important changes take place in the biosphere. The repository sites are expected to be covered by ice sheets or sea during long periods, and the aggregate effect of climate change will therefore be a reduction of the dose consequences compared with a situation where the present-day climate persists.

Earthquake scenario

In the earthquake scenario, the consequences of earthquakes are analyzed by means of model studies where site-specific data are used for the structure of the geosphere and for earthquake statistics. The analysis method is new and includes several highly pessimistic simplifications. The analyses show that the probability of canister damage is comparable with the probability assumed for initial damage in the canister defect scenario. In the evaluation of the analysis method, it is shown how less pessimistic assumptions should lead to no canister damage at all in the model studies. The method will be refined.

Intrusion scenario

The scenario that deals with future inadvertent human actions that could conceivably affect the repository is surrounded by great uncertainties, chiefly because the evolution of human society is in principle unpredictable. SR 97 discusses how conceivable societal evolutions and future human actions that affect the repository can nevertheless be categorized to some extent. In an illustrative example, a situation is analyzed where a canister in the repository is inadvertently penetrated by rock drillers. Dose and risk are calculated for the drilling personnel and for a family that settles on the site at a later point in time. The risk to both drilling personnel and family is judged to lie well below the acceptance criterion, since the probability of the analyzed events is estimated to be very small.

Conclusions

The principal conclusion of the SR 97 safety assessment is that the prospects of building a safety deep repository for spent nuclear fuel in Swedish granitic bedrock are very good.

The three analyzed sites reflect reasonable variations of the conditions in granitic bedrock in Sweden. The analysis does not provide support for attaching any significant importance to differences in long-term safety between sites in a weighing together of all the factors that influence the siting of a deep repository.

Another conclusion is that the methodology that is used in SR 97 comprises a good foundation for future safety assessments that will be based on data from completed site investigations.

The results of the assessment also serve as a basis for formulating requirements and preferences regarding the bedrock in site investigations, for designing a programme for site investigations, for formulating functional requirements on the repository's barriers, and for prioritization of research.

The next stage in the siting of a deep repository entails investigation of the bedrock at a number of candidate sites in Sweden. It is SKB's judgement that the scope of the safety assessment and confidence in its results satisfy the requirements that should be made in preparation for such a stage.

1 Premises

Under Swedish law, the owners of nuclear reactors are obligated to see to it that radioactive waste from their activities is managed and disposed of safely. The Swedish power utilities jointly own Svensk Kärnbränslehantering AB, SKB (the Swedish Nuclear Fuel and Waste Management Company), whose mission is to develop methods for managing radioactive waste and to build and operate the facilities required for this.

Spent nuclear fuel is an important component in the radioactive waste, since it is both highly radioactive (high-level) and long-lived. At present, spent fuel is stored for a year or so at the reactor, after which it is transferred to CLAB, a central interim storage facility for spent nuclear fuel. According to SKB's plans, after 30 to 40 years of interim storage the fuel will be encapsulated in copper canisters and disposed of at a depth of approximately 500 metres in the crystalline bedrock. The facilities required for this, an encapsulation plant and a deep repository, have not yet been sited and built.

The system will be constructed over a period of several decades. Siting of facilities and systems is done in collaboration with concerned municipalities and under the supervision of safety and radiation protection authorities, all subject to the approval of the Government.

1.1 Why SR 97?

In preparation for the next stages in the realization of the system, the Swedish Government stated the following in its decision following the review of SKB's research programme RD&D 95:

"A safety assessment of the repository's long-term safety should, in the opinion of the Government, be completed before an application for a permit to construct an encapsulation plant is submitted, likewise before site investigations on two or more sites are commenced."

This report gives an account of the requested safety assessment before site investigations are commenced. The working title of the analysis is SR 97 (Safety Report 97).

In its review of SKB's RD&D 98, the Swedish Nuclear Power Inspectorate (SKI) clarifies its view of the purpose and requirements for SR 97:

"The purpose is to demonstrate that the KBS-3 method has good prospects of being able to meet the safety and radiation protection requirements which SKI and SSI have specified in recent years."

SKI also writes: "…that SR 97, besides demonstrating a methodology for safety assessment, should also serve as a basis for:

- demonstrating the feasibility of finding a site in Swedish bedrock which meets the requirements on long-term safety and radiation protection that are defined in SSI's and SKI's regulations,

- specifying the factors that serve as a basis for the selection of areas for site investigations,
- deriving which parameters need to be determined and which other requirements ought to be made on a site investigation,
- deriving preliminary functional requirements on the canister and the other barriers."

1.2 Purposes

Based on the above points, four concrete purposes for SR 97 can be formulated:

1. SR 97 shall serve as a basis for demonstrating the feasibility of finding a site in Swedish bedrock where the KBS-3 method for deep disposal of spent nuclear fuel meets the requirements on long-term safety and radiation protection that are defined in SSI's and SKI's regulations.

2. SR 97 shall demonstrate methodology for safety assessment.

The ambition of SR 97 is to carry out a complete analysis of the long-term safety of the KBS-3 system for deep disposal of spent nuclear fuel. The methodology employed in SR 97 includes:

- a systematic handling of all the internal processes and external conditions that can cause long-term changes in the repository, and
- a systematic handling of the different types of uncertainties that always surround the background data for an analysis.

SR 97 is based on data from three actual sites. Data have been taken from SKB's investigations at Gideå in Ångermanland, from Finnsjön in northern Uppland County and from the Hard Rock Laboratory on Äspö outside Oskarshamn in Småland. The sites have been selected as calculations examples to reflect different conditions in Swedish granitic bedrock as regards geology, groundwater flux, water chemistry, nearness to coast, northerly or southerly location, surrounding biosphere, etc.

The report on the execution and results of the analysis therefore serves as a direct basis for assessing: a) the feasibility of finding a safe site for a KBS-3 repository in Swedish bedrock, and b) the methodology for a safety assessment.

3. SR 97 shall serve as a basis for specifying the factors that serve as a basis for the selection of areas for site investigations and deriving which parameters need to be determined and which other requirements ought to be made on a site investigation.

SR 97 comprises an important supporting document in the ongoing work of formulating requirements and preferences regarding the rock from the perspective of longterm safety. Results and experience from SR 97 are also used directly in the work of formulating an integrated programme for investigations and evaluations of sites. The conclusion chapter summarizes the way in which SR 97 comprises a background document for these two efforts.

4. SR 97 shall serve as a basis for deriving preliminary functional requirements on the canister and the other barriers.

How functional requirements can be derived from the results of the safety assessment is discussed in the conclusion chapter.

1.3 Delimitations

SR 97 is a complete safety assessment of the KBS-3 method for deep disposal of spent nuclear fuel, where geosphere data are taken from three actual sites in Sweden. The following fundamental premises also apply:

Post-closure safety

SR 97 deals with the long-term safety of the repository after closure. The construction and operating phases are not dealt with. These phases, as well as other aspects that pertain to the whole waste management system (encapsulation, transportation and deep disposal), are described in preliminary safety reports in conjunction with construction and operation. Together with SR 97, they comprise the background material for an integrated system analysis of all components in the waste management system to be published in 2000. Nor is SR 97 concerned with safety in connection with a prolonged open period or a partially closed repository.

Repository for spent nuclear fuel

SR 97 is concerned with a repository for spent nuclear fuel. Other long-lived waste will also have to be disposed of, for example core components from the decommissioning of nuclear power plants and waste from previous activities at the research reactor at Studsvik. This waste will be emplaced in a separate repository, which can be co-sited with the repository for spent nuclear fuel or with the final repository for radioactive operational waste, SFR, which is in operation today. The repository can also be sited separately.

A preliminary facility design and safety assessment for such a repository has been prepared in parallel with SR 97 and is presented in a separate report /SKB, 1999/. The safety-related consequences of a possible co-siting are not investigated in SR 97, but both assessments are based on the same geological data.

Holistic view of radiation protection

Long-term post-closure safety is one aspect of a holistic view of radiation protection in connection with waste management. A complete picture is presented in the system analysis mentioned above. The options allowed within the frames for KBS-3, as well as when and based on what information they will be evaluated and screened, are also discussed in that report.

1.4 Report structure

The structure of the account in SR 97 represents a development of the template devised in 1995 in safety report SR 95.

The body of material for a safety assessment is very large. SR 97 is presented in the form of a main report to which three main references are closely associated, see Figure 1-1. In the main report and the three main references, reference is made to reports in SKB's report series or in the open literature.

The main report – ”Deep Repository for Spent Nuclear Fuel; SR 97 – Post-closure safety” – summarizes the entire safety assessment. It can be read separately from the others and includes methodology description, all essential results, as well as evaluations and conclusions. The report consists of two parts and a summary (this volume). All parts are available in Swedish and English.

"SR 97 – Waste, repository design and sites" describes in detail the waste, the repository design with canisters and buffer/backfill material, the three sites and the site-specific adaptations of the repository layouts that have been done. The report is available in both Swedish and English. Hereinafter, this report will be referred to as the "Repository System Report".

"SR 97 – Processes in the repository evolution" describes the thermal, hydraulic, mechanical and chemical processes in fuel, canister, buffer and geosphere that control the evolution of the repository system. The report is available in both Swedish and English. Hereinafter, this report will be referred to as the "Process Report".

"SR 97 – Data and data uncertainties" (in English only) contains a compilation of input data for calculations of radionuclide transport. There is also an evaluation of uncertainties in input data. Hereinafter, this report will be referred to as the "Data Report".

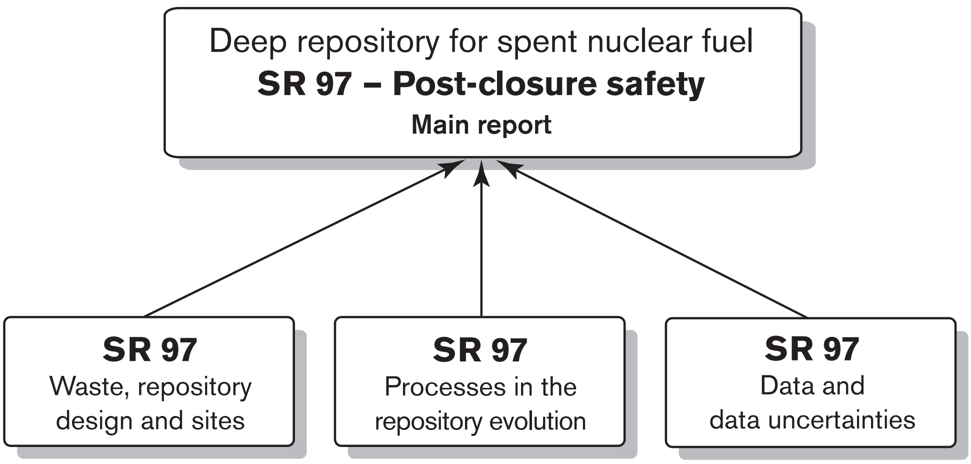

Figure 1-1. *Main report and some of the most important references to SR 97.*

1.5 Acceptance criteria

The form and content of a safety assessment, and above all the criteria for judging the safety of the repository, are defined in regulations issued by the Swedish safety and radiation protection authorities. The regulations are based on framework legislation, the most important being the Environmental Code, the Nuclear Activities Act and the Radiation Protection Act. Radiation protection matters are handled by a number of international bodies, and national legislation is often based on international rules and recommendations.

Long-term safety is regulated today by the Swedish Radiation Protection Institute's (SSI) "Regulations for final disposal of spent nuclear fuel" (SSI FS 1998:1). The regulations entered into force on 1 February, 1999.

In 1999, the Swedish Nuclear Power Inspectorate, SKI, distributed a draft version of "The Swedish Nuclear Power Inspectorate's regulations concerning safety in final disposal of nuclear waste".

1.5.1 SSI's regulations for final disposal of spent nuclear fuel

SSI writes that human health and the environment, now and in the future, shall be protected from the harmful effects of ionizing radiation. Nuclear activities must not cause more serious effects on human health and the environment outside Sweden's boundaries that what is acceptable within Sweden. A final repository shall be designed so that no additional measures are needed after closure to prevent or limit the escape of radioactive substances from the repository. Institutional controls and knowledge of the location of the repository in a distant future cannot be assumed. SSI's regulations apply to the long-term safety of a closed repository.

Protection of human health

The overall acceptance criterion for a deep repository is expressed in Section 5 of SSI's regulations:

"A final repository for spent nuclear fuel or nuclear waste shall be designed so that the annual risk of harmful effects after closure is no more than 10^{-6} for a representative individual in a group that is exposed to the greatest risk."

The acceptance criterion is thus a risk measure. A risk calculation investigates what courses of events can lead to harmful effects, what their probability of occurring is, and the size of the injury (the consequence) for each course of events. The product of probability and consequence gives a sub-risk for each course of events. The aggregate risk is the sum of the sub-risks for different conceivable courses of events.

SSI stipulates an annual risk of 10^{-6} for individuals exposed to radiation from the repository. For a hypothetical situation with exposure that occurs with certainty (probability = 1), this corresponds to an annual radiation dose of 0.015 milliSieverts (mSv) from the repository. This can be compared with the natural background radiation, which is several mSv/y in Sweden.

The risk limit applies to a representative individual in the group that is exposed to the greatest risk. As an indication of the size of such a group, SSI mentions the population in an area where it is theoretically possible to site ten different deep repositories. Such an area is difficult to delimit in a risk calculation. As an alternative, SSI states that it "can be acceptable to carry out the calculations for an individual judged to be highly burdened, instead of an individual who is representative for the whole group's burden".

The risk limit for such an individual is set at 10^{-5}, which corresponds to a radiation dose of 0.15 mSv/y. The exposure models in SR 97 have not been adapted to the details of SSI's regulations, since the latter did not enter into force until towards the end of the assessment. However, the models are already designed in most cases to calculate doses to a small and highly exposed group which, for example, lives solely on contaminated food. The calculation result in SR 97 should therefore in most cases be compared with the risk criterion 10^{-5}/y, equivalent to a dose limit of 0.15 mSv/y for an exposure that occurs with certainty, see further section 3.3.7.

Environmental protection

SSI also states that:

"§7 An account shall be given of biological effects of ionizing radiation in affected habitats and ecosystems. The account shall be based on available knowledge of concerned ecosystems …"

In the absence of established methodology, SSI says that the precautionary principle shall apply, i.e. the very suspicion of harmful effects on the environment shall be sufficient to intervene or refrain from a given activity.

In SR 97, the biological effects of a release are judged by comparison with the natural background radiation. If the releases are small compared with the background radiation, the effects should be negligible.

Intrusion

SSI stipulates that an account shall be given of the consequences of an inadvertent intrusion or other disturbance in the final repository or its vicinity. What is essential is not to describe the chain of events leading up to the intrusion, but to shed light on the repository's protective function after an intrusion. The protective capacity of a final repository must not be impaired by planned measures to hinder intrusion or facilitate retrievability.

Doses higher than 1 mSv/y, which could conceivably be encountered in connection with an intrusion into the final repository, will be assessed separately by SSI.

Time periods

SSI states that harmful effects in the future should not be regarded as less important than the harmful effects to which man or the environment are exposed today.

SSI emphasizes that the first 1,000 years after repository closure is the most important period to investigate, since the radiotoxicity of the waste is greatest then. The highest demands are made on the safety account for this period. The regulations also require an account of a case based on the assumption that the biosphere conditions prevailing at the time of the licence application do not change. The term "prevailing conditions" also takes into account known changes such as postglacial land uplift.

The period after the initial 1,000 years shall also be investigated, and SSI emphasizes the importance of accounting for the different types of uncertainties in the underlying data on which the analyses of different epochs are based.

1.5.2 SKI's draft version of regulations concerning safety in final disposal of nuclear waste

The regulations from SKI are as yet only available in a draft version. Among other things, the regulations talk about how the safety assessment should deal with various internal and external conditions that may have a bearing on safety. SKI emphasizes the importance of a systematic handling of uncertainties, and that the models and data used should be demonstrated to be applicable as far as possible. The assessment must cover the first million years after repository closure.

Since the regulations are not yet available in a final version, it has not been possible to use them as a direct basis for SR 97. In general, it can nonetheless be said that all aspects dealt with in the draft version are also covered in one way or another in SR 97.

1.6 Safety principles

As the work of developing a safe deep repository in Sweden has proceeded, a philosophy has emerged regarding how the radioactive waste in Sweden is to be managed. In brief, it entails the following:

- Long-term safety shall not require future monitoring and maintenance.
- The repository shall be designed to permit possible future measures to modify the repository or retrieve the waste.
- The long-term safety of the repository shall be based on multiple engineered and natural barriers which contribute via different functions to the repository's total safety.

The practical application of this philosophy has resulted in a repository design with a multiple barrier system, the KBS-3 system.

The KBS-3 repository for spent nuclear fuel is designed primarily to isolate the waste. If the isolation function should for any reason fail in any respect, a secondary purpose of the repository is to retard the release of radionuclides. This safety is achieved with a system of barriers that support and complement each other. The safety of the repository must be adequate even if one barrier should be defective or fail to perform as intended. This is the essence of the multiple barrier principle.

Another principle is to make the repository "nature-like", i.e. to use natural materials for the engineered barriers. Choosing materials from nature makes it possible to judge and evaluate the materials' long-term stability and behaviour in a deep repository based on knowledge of natural deposits. For the same reason, the repository should cause as little disturbance of the natural conditions in the rock as possible. Above all, an attempt is made to limit the chemical impact of the repository in the rock.

1.7 Time perspective

The repository should function as long as the waste is hazardous. It takes many billions of years before all radioactive material has decayed to stable elements. By then, however, their radiotoxicity has long since declined to levels comparable to the radiotoxicity of the uranium ore originally mined to produce the fuel.

Approximately eight tonnes of natural uranium are enriched to fabricate one tonne of fuel for a Swedish reactor. During reactor operations, the radiotoxicity of the fuel increases as new radioactive substances are formed when uranium nuclei undergo fission. Figure 1-2 shows how the radiotoxicity of the spent fuel subsequently declines with time. After approximately 100,000 years, the radiotoxicity of a tonne of spent fuel is on a par with that of the eight tonnes of natural uranium used in fabricating the fuel.

The figure 100,000 years can therefore be used as a guideline for how long the repository has to "function". However, this figure is not an absolute time limit in the evaluation of the repository's safety:

- On the one hand, radiotoxicity declines steadily and has e.g. after a thousand years fallen to about one-tenth of the level at deposition. This is important in the evaluation of the repository's safety: Uncertainty regarding conditions in and around the repository grows with time, but at the same time the radiotoxicity of the fuel diminishes.
- On the other hand, even after 100,000 years there are both small quantities of radionuclides that can move relatively easily through the repository's barriers if the copper canister should be damaged, and larger quantities of low-mobility nuclides.

The safety of the repository thus needs to be evaluated far into the future and constantly in the light of how radiotoxicity declines with time.

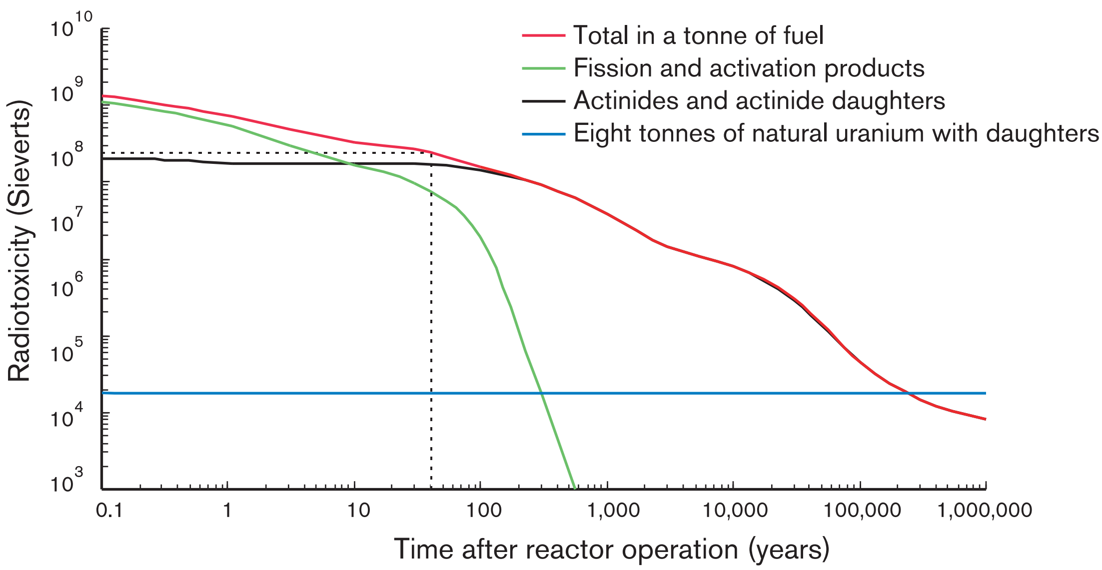

Figure 1-2. *Toxicity of the waste as a function of time after discharge from the reactor for Swedish BWR fuel with a burnup of 38 MWd/t U. Radiotoxicity pertains to ingestion via food. After 30 to 40 years of interim storage, the fuel will be deposited in the final repository.*

1.7.1 Time periods in SR 97

SR 97 deals with a closed repository. In practice, the repository will be filled in stages over a period of approximately 50 years, but in the analysis it is assumed that all fuel is deposited at the same time. Prior to disposal, it is assumed that the fuel has been kept in interim storage for 40 years after discharge from the reactor.

It is assumed in SR 97 that there are institutional controls for the first one hundred years, which means that inadvertent intrusion can be ruled out during this period.

In accordance with SKI's above proposed regulations, one million years is the upper time limit for the analyses in SR 97.

1.8 Methodology

1.8.1 What is a safety assessment?

Safety assessment is the method that is used to analyze and judge the performance and safety of a final repository in a systematic manner. A safety assessment of a deep repository can in simple terms be said to consist of the following tasks:

- carefully describe the appearance or state of the repository system when it has just been closed,
- survey what changes the repository could conceivably undergo in time as a consequence of both internal processes within the repository and external forces,
- evaluate the consequences of the changes for safety.

In both the execution and the presentation of SR 97, the processes are in focus. Knowledge of all known internal processes of importance for long-term safety is documented in a special report, the Process Report. The processes and their couplings to each other are illustrated schematically in THMC diagrams, where the processes are classified into the categories thermal (T), hydraulic (H), mechanical (M) and chemical (C). In the analysis, groups of coupled processes are linked together into a description of an integrated evolution in time.

The execution and presentation of SR 97 can be divided into five steps:

1. System description

A systematic analysis requires a structured description of the repository and of all the internal processes, their interrelationships and the properties of the repository that are influenced by a particular process. Preparing such a system description is therefore the first task in a safety assessment. This task also includes defining the boundary between a system and its surroundings. The THMC structure is used for the system description in SR 97. The methodology is described more fully in section 2.1.

2. Description of initial state

The initial state of the repository, i.e. what it looks like when it has just been closed, is then described. This includes a description of the dimensions and materials in the engineered barriers as well as the properties of the bedrock around the repository as they appear initially. Gaps in our knowledge of the initial state are also described.

3. Choice of scenarios

The evolution of the repository is influenced by its surroundings. Assessments of the evolution of the surroundings necessarily contain uncertainties: What climatic conditions can be expected in the future? What frequencies and magnitudes of earthquakes can be expected in the repository's surroundings in the future? To cover different situations in the surroundings, the evolution of the repository is analyzed for a number of different sequences of events in the surroundings: a number of different scenarios are selected and analyzed. The chosen scenarios should together provide reasonable coverage of the different evolutionary pathways the repository and its surroundings could conceivably take.

Scenarios can also be based on different assumptions regarding the initial state in the repository of importance for its long-term evolution.

4. Analysis of chosen scenarios

With the aid of the system description, the evolution of the repository is analyzed for each scenario. A number of different tools and methods are used here, ranging from reasoning and simple approximations to detailed modelling based on site-specific data.

In SR 97, a base scenario is first analyzed where the repository is postulated to be built according to specifications and where present-day conditions in the surroundings, including climate, are assumed to persist.

A number of other scenarios are then analyzed where the course of events is compared with that in the base scenario. How will the evolution of the repository change if the climate changes? In the event of earthquakes? If a barrier has a fabrication defect? What importance do these changes have for safety? A basis for carrying out the analyses and reporting them in the form of comparisons with a base scenario is that the analyzed repository system is engineered to be robust, i.e. so that varying conditions in the surroundings do not cause dramatic changes in the evolution and performance of the repository.

5. Evaluation

Finally, an overall assessment is made of repository safety, where the different scenarios are weighed together into a total risk picture. The conclusions of the overall assessment comprise the results of the safety assessment. Confidence in the results in the light of the uncertainties that exist in the data underlying the assessment is also discussed here.

1.8.2 Handling of uncertainties

Just as important as the assessment of the repository's protective capacity is confidence in the results. The background data for a safety assessment is always burdened with different types of deficiencies. It is, for example, never possible to know in detail the fracture structure in the host rock, or to achieve certainty about the future climate. Repository safety must be evaluated in the light of such deficiencies. To put it simply, we are faced with the task of showing whether the repository has been designed with adequate margins to be safe in spite of inadequate knowledge. Confidence in the results is dependent on, among other things, how methodically uncertainties and deficiencies have been handled.

Handling of uncertainties is not a separate activity, but comprises an integral part of the analysis work. Uncertainty handling is nevertheless discussed independently here, since it comprises an important part of the methodology.

Deficiencies can be of a qualitative or quantitative nature. Qualitative deficiencies concern e.g. questions of completeness: Have all processes that influence the evolution of the repository been identified in the system description? Have all types of external impact been covered in the choice of scenarios? Other qualitative questions concern process understanding: Do we understand the internal processes well enough for the needs of the safety assessment? Do we understand the processes that determine conditions in the surroundings well enough?

Other questions are quantitative. How well can the initial state be determined? The initial temperature of the repository can be determined with an accuracy which is fully adequate for the needs of the analysis, while the description of fracture geometry in the geosphere is burdened with uncertainties that require more careful handling. How well can be describe different processes quantitatively, for example heat conduction or groundwater flow? This question is particularly important for the analysis of radionuclide transport, which is of direct importance to the evaluation of the repository's safety. Calculations of radionuclide transport handle large amounts of input data, which may be burdened with varying degrees of uncertainty.

Handling of uncertainties consists of both reporting uncertainties and deficiencies in the underlying data and handling them when conducting the analysis. Table 1-1 shows the different types of uncertainties which enter into the first four steps of the analysis.

As is evident from the table, what is often termed "conceptual uncertainty", with varying implications in various contexts, has in SR 97 been divided into the concepts "fundamental process understanding" and "model uncertainty". The former refers to the scientific understanding of a process, while the latter refers to the uncertainties which arise when a process is described with the aid of a model in a safety assessment. Our fundamental understanding of the process of water flow is good, for example. Mathematical models of groundwater movements in fractured rock can nevertheless be formulated in different ways, which entails an uncertainty. The different formulations represent different ways of handling the fact that the details in the nature of the fracture system are only known in a statistical sense.

Uncertainties in data for radionuclide transport

The results of the radionuclide transport calculations quantify repository safety. It is thereby particularly important to assure the quality of these calculations. The handling of uncertainties in input data to radionuclide transport calculations must therefore be rigorous.

Data for transport calculations in SR 97 are taken from background reports by experts within the relevant field. Many of the reports have been specially produced for SR 97. Each author has discussed uncertainties in data according to a given template. The

Table 1-1. The types of uncertainties that enter into the first four steps of the analysis and where these are presented and handled in the main report summary.

1. System description	
Completeness	Discussed in section 2.7.
Process understanding (conceptual uncertainty)	Described in detail in a separate report, the Process Report.
2. Description of initial state	
Data uncertainty	Quantitative uncertainties in the initial state are described for each variable in the complete main report, Chapter 6.
3. Choice of scenarios	
Completeness/Coverage	Discussed in section 3.1.1.
4. Analysis of chosen scenarios	
Model uncertainty (conceptual uncertainty)	General confidence in models used is discussed briefly where model studies are reported in Chapter 3. Conficence in models for radionuclide transport and groundwater flow is discussed more thoroughly in the complete main report, section 9.11.
Input data to models/ calculations	General data uncertainties are discussed briefly where calculations and model studies are reported in Chapter 3. Data uncertainties that concern radionuclide transport and groundwater flow are reported and discussed in detail in a special Data Report. Uncertainty analyses and probabilistic analyses are done in the calculations of radionuclide transport.

discussion of uncertainties in the individual reports has then been evaluated in the Data Report. This document has also proposed for all input data:

- a reasonable value and
- a pessimistic value.

In addition to a reasonable and a pessimistic value, statistical distributions have also been presented for some input data, such as data that exhibit spatial variability or data determined by spatially variable conditions. In the geosphere, the heterogeneity in e.g. the fracture system gives rise to a distribution in space of various data for calculations, often after extensive analyses.

The span between reasonable and pessimistic values offers an option for representing uncertainties in input data. Radionuclide transport calculations are carried out with different combinations of reasonable and pessimistic data in order to shed light on the importance of different uncertainties.

Furthermore, probabilistic analyses of some kind are required to arrive at a risk measure that can be compared with the acceptance criterion for the deep repository. The following approach is used in SR 97 for the probabilistic analyses:

- Statistical distributions are used only where there is some kind of statistical material on which to base a distribution, see the examples above.
- Reasonable and pessimistic values are used for other data. Both are assigned probabilities that are chosen so that the risk is deemed to be overestimated in the probabilistic analysis.

2 System description/initial state

The KBS-3 repository for spent nuclear fuel is composed of a system of barriers, see Figure 2-1:

- The fuel is placed in corrosion-resistant copper canisters. Inside the five-metre-long canisters is a cast iron insert that provides the necessary mechanical strength.
- The canisters are surrounded by a layer of bentonite clay that protects the canister mechanically in the event of small rock movements and prevents groundwater and corrosive substances from reaching the canister. The clay also effectively adsorbs radionuclides that are released if the canisters should be damaged.
- The canisters with surrounding bentonite clay are emplaced at a depth of about 500 metres in the crystalline bedrock, where mechanical and chemical conditions are extremely stable.
- If any canister should be damaged, the chemical properties of the fuel and the radioactive materials, for example their poor solubility in water, put severe limitations on the transport of radionuclides from the repository to the ground surface. This is particularly true of those elements with the highest long-term radiotoxicity, such as americium and plutonium.

The safety assessment is based on the repository system as it appears at an initial point in time, which in SR 97 is at closure. The way in which the repository system is changed by internal processes and external forces is then analyzed.

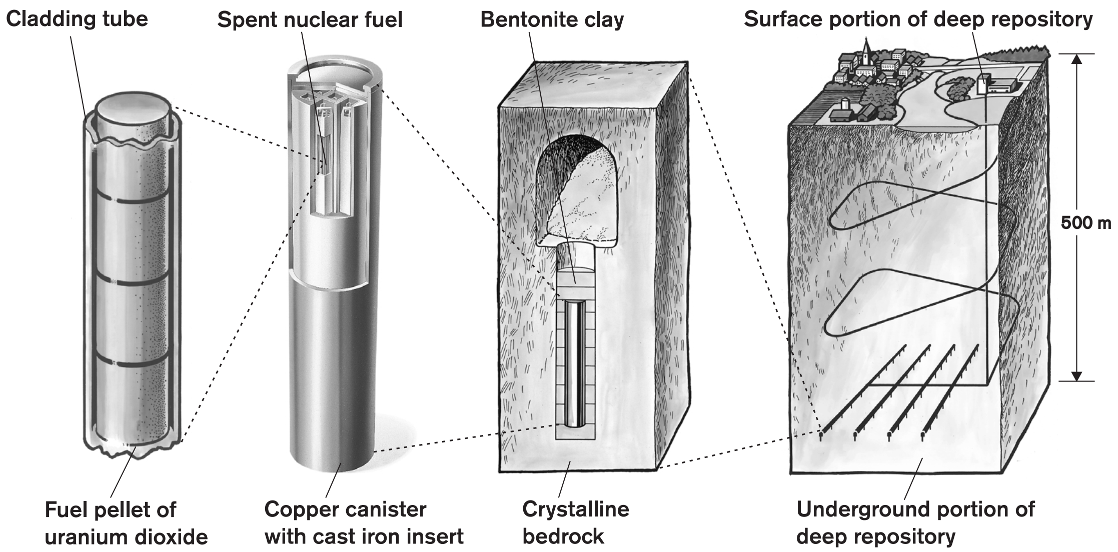

Figure 2-1. The KBS-3 system.

A complete analysis requires a formalized description of the repository system and of the processes, with all their couplings and relationships, that control its evolution. The methodology used in SR 97 entails first describing processes, variables and couplings schematically in a system description, after which the variables are assigned initial values, the formalized description of the initial state.

This methodology is applied strictly in the complete version of the main report. In this summary, a simplified account of the methodology for system description is first provided. After that, the initial state of the different barriers is described in greater detail, but in a manner not as strictly formalized as in the complete main report. The description of the geological barrier, the host rock, is at the same time site descriptions of the three sites analyzed in SR 97.

At the end of the chapter, a number of technical criteria are set up against which long-term safety can be evaluated, followed by a discussion of the completeness of the system description.

2.1 Methodology

2.1.1 System description of THMC format

A systematic analysis requires a description of all known internal processes of any conceivable importance, their interrelationships and the properties of the repository that are influenced by the particular process. The structure of the description should provide both an overview and details. Another requirement on the structure is that it must be able to be used throughout in the presentation of the safety assessment.

For the system description, the repository is divided into the four subsystems fuel, canister, buffer/backfill and geosphere. All known thermal, hydraulic, mechanical and chemical processes that are of importance for the evolution of the repository are identified for each subsystem. Influences between subsystems, such as buffer and geosphere, are also charted. These influences are also primarily thermal, hydraulic, mechanical or chemical by nature. This method of structuring processes and interactions in the safety assessment is new for SR 97, while the work of identifying the processes included in the structure has been going on for some time.

Variables determine state

The state in a subsystem is characterized at any given moment by a set of variables. The state of the geosphere is, for example, characterized by its temperature, which varies in time and space, by its fracture geometry (which varies widely in space, but hardly at all in time), by groundwater flow, groundwater composition, rock stresses, etc.

Together, the variables should characterize the system sufficiently well to enable a safety assessment to be conducted. Some variables, such as temperature and groundwater composition, are used or determined directly in analyses and calculations, while others serve as a basis for deriving important properties of the system: The thermal conductivity and density of the geosphere can, for example, theoretically be calculated from the variable matrix minerals.

All variables are time-dependent and are influenced by one or more processes, and all processes are influenced by one or more variables.

THMC diagram

All the processes and variables for each subsystem and their interdependencies are gathered into a diagram, which also includes interactions with adjacent subsystems. The diagram is called a THMC diagram, after the classification of the processes and interactions into thermal (T), hydraulic (H), mechanical (M) and chemical (C) categories. The THMC diagram for buffer/backfill is shown in Figure 2-2. The diagram also contains radiation-related processes, which have to do with radioactive decay and radiation attenuation in the repository system and processes related to the transport of radionuclides.

Documentation of processes

The system description also includes documentation of knowledge concerning each process. The process documentation constitutes a cornerstone in the background material for a safety assessment. Knowledge of all identified processes in the system description is documented for SR 97 in the Process Report. The following are given for each process:

- general description of the process,
- documentation of model studies/experimental studies,
- discussion of uncertainties in both understanding and data for the process,
- proposals as to how the process can be handled for different scenarios in the safety assessment.

2.2 Fuel

The total quantity of fuel obtained from the 12 Swedish nuclear reactors will depend on operating time, energy output and fuel burnup. As of the beginning of 1998, approximately 4,000 tonnes of spent fuel had been generated. With an operating time of 40 years for all reactors, the total quantity of spent fuel can be estimated at 9,500 tonnes. The equivalent quantity for 25 years' operating time is 6,500 tonnes.

In SR 97 it is assumed that approximately 8,000 tonnes of fuel will be disposed of. It is assumed for the sake of simplification in most subanalyses that all canisters contain fuel from boiling water reactors, BWR fuel, of type SVEA 96 with a burnup of 38 MWd/tU.

Structure of the fuel assemblies

Nuclear fuel consists of cylindrical pellets of uranium dioxide. The pellets are 11 mm high and have a diameter of 8 mm. In SVEA 96 fuel, the pellets are stacked in approximately 4-metre-long cladding tubes of Zircaloy, a durable zirconium alloy. The tubes, or "cans", are sealed with welds and bundled together into fuel assemblies. Each assembly contains 96 cladding tubes. A fuel assembly also contains components of the nickel alloys Inconel and Incoloy, and parts of stainless steel.

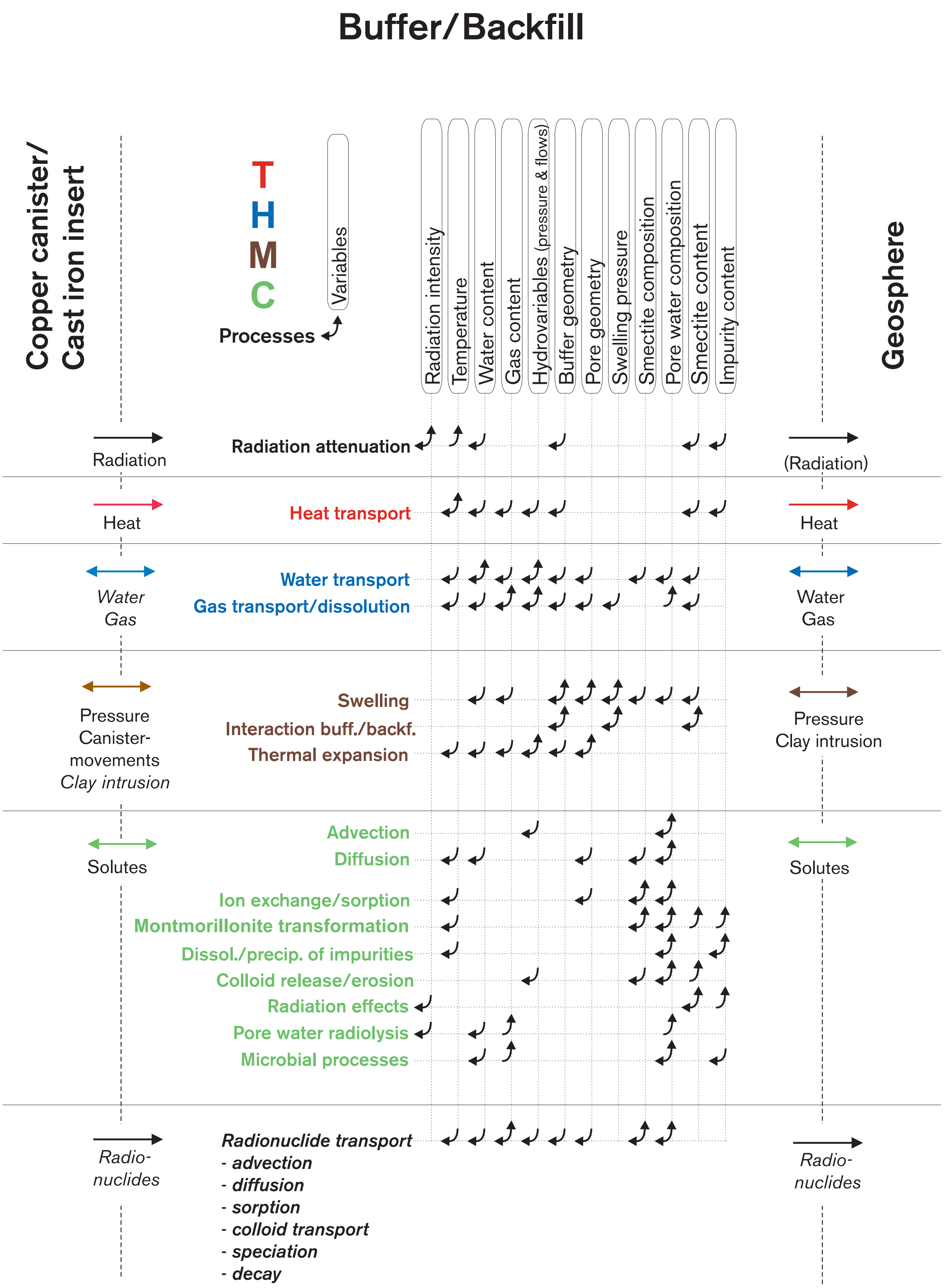

Figure 2-2. THMC diagram for buffer/backfill.

Radionuclides

Radionuclides are formed during reactor operation by nuclear fission of uranium-235 and plutonium-239 in particular, and by neutron capture by nuclei in the metal parts of the fuel. The former are called fission products, the latter activation products. Moreover, uranium can form plutonium and other heavier elements by absorbing one or more neutrons. Most of the radionuclides lie embedded in the fuel matrix of uranium dioxide. A few fission products are relatively mobile in the fuel and may migrate to the surface of the fuel pellets during operation.

THMC description of processes and variables

In the complete THMC description of the fuel, important variables are e.g. quantity and distribution of radionuclides, as well as dimensions and material composition of above all the fuel matrix and the cladding tubes. Important fuel-related processes are fuel dissolution and dissolution/precipitation of radionuclides.

2.3 Cast iron insert/copper canister

The canister consists of an inner container of cast iron and a shell of copper. The cast iron insert provides mechanical stability and the copper shell protects against corrosion in the repository environment. The copper shell is 5 cm thick and the canister takes the form of an approximately 4.8 metre tall cylinder with a diameter of 1.05 metres.

The insert has channels where the fuel assemblies are placed and is available in two versions: one for 12 BWR assemblies and one for 4 PWR assemblies. The fuel channels are fabricated in the form of an array of square tubes. The walls and bottom of the inner container are then fabricated by pouring spheroidal graphite iron around the channel array.

The copper canister is fabricated either of drawn seamless tubes or by welding together two tube halves of rolled plate. A bottom is attached by an electron beam weld in such a way that the weld can be examined by ultrasonic and radiographic inspection.

After fuel has been placed in the canister, the copper shell's lid is then attached by an electron beam weld, and leaktightness is tested by ultrasonic and radiographic inspection.

The canister weighs a total of about 25 tonnes when filled with 12 BWR assemblies. A canister holds about two tonnes of fuel. It is assumed in SR 97 that approximately 8,000 tonnes of fuel will be disposed of, equivalent to around 4,000 canisters.

THMC description of processes and variables

In the complete THMC description of the canister, important variables are e.g. material composition and dimensions of copper shell and cast iron insert, which determine the strength and corrosion resistance of the canister. Important processes in the canister in the long term are copper corrosion and deformation under external loading.

2.4 Buffer/backfill

In their deposition holes, the copper canisters will be surrounded by a buffer of bentonite clay. On deposition, gaps are left for technical reasons between canister and buffer and between buffer and rock. The inner gap is filled with water and the outer with bentonite pellets and water.

After deposition, the tunnel above the deposition hole will be backfilled with a material that is adapted above all to groundwater salinity on the repository site.

Buffer material

The buffer consists of MX-80 bentonite, a natural clay from Wyoming or South Dakota in the USA. The designation MX-80 is a trade name and specifies a certain grade and particle size of dried and ground bentonite.

MX-80 bentonite consists mainly of the smectite mineral montmorillonite (65–80 percent), where the clay particles are smaller than 2 µm. Chemically, montmorillonite can be described as a polyelectrolyte, where exchangeable ions are associated with the surfaces of the negatively charged clay particles. A characteristic property of the clay is that it swells in contact with water. The exchangeable ions in MX-80 consist predominantly of sodium, and the material is therefore called sodium bentonite. Water-saturated buffer contains about 25 weight-percent water. The water molecules are absorbed in the material, and water transport takes place chiefly by diffusion.

MX-80 bentonite also contains the minerals quartz (about 15 percent) and feldspar (5–8 percent). Chemically important components in addition to the minerals are carbonates (e.g. calcite), sulphates, fluorides, sulphides (e.g. pyrite), iron(II) and organic matter.

After wetting, the bentonite will contain a pore water of a characteristic composition that depends on the composition of the bentonite and of the water used for wetting.

Backfill material

The backfill material consists of a mixture of bentonite clay and crushed rock. The proportions are adapted to the chemical conditions on the repository site so that the backfill will have the desired characteristics. Such site-specific adaptation has not been done in SR 97. Instead, a typical composition is used consisting of 15 weight-percent MX-80 bentonite clay and 85 weight-percent crushed rock.

THMC description of processes and variables

In the complete THMC description of buffer/backfill, important variables are e.g. material composition, dimensions, water content and density. Density in particular determines the capacity of the buffer to support the canister. Material composition and water content determine the buffer's capacity to limit transport of dissolved corrodants in to the canister, and of any radionuclides to surrounding rock. Important buffer-related processes are water uptake, swelling and chemical transport and reaction processes.

2.5 Geosphere/site descriptions

2.5.1 Crystalline rock

The deep repository will be situated in crystalline rock of granitic composition. Granitic bedrock consists for the most part of quartz, feldspars, mica minerals and amphiboles (hornblende). In addition there are small quantities of accessory minerals, which may be of geochemical importance.

The crystalline rock is also characterized by a system of fractures. The frequency, spatial distribution, size distribution, shape and orientation of the fractures are crucial in determining both hydraulic and mechanical properties in the rock. Fractures occur on all scales from microscopic fractures in the rock matrix to fracture zones, i.e. large zones of significantly elevated fracture frequency in relation to the surrounding rock. Fracture zones often constitute dominant flow paths for the groundwater, and the size of the rock movements that can occur in a fracture zone is related to the extent of the zone.

2.5.2 The three sites in SR 97

Three hypothetical repository sites are analyzed in SR 97 to illustrate actual conditions in Swedish crystalline bedrock. Data are taken from Äspö in Småland, Finnsjön in Uppland and Gideå in Ångermanland, see Figure 2-3. The sites represent three areas in stable geological settings. All three sites are relatively near the coast, and Äspö is an offshore island.

The three sites have been investigated by different experts on different occasions over a twenty-year period in studies of somewhat differing purpose and scope. The quantity of material is biggest for Äspö and smallest for Gideå.

The repositories that are analyzed on the three sites are hypothetical. None of the sites are being considered for site investigations in the ongoing siting work. To emphasize this, the names Aberg, Beberg and Ceberg will be used from now on for the sites at Äspö, Finnsjön and Gideå, respectively.

Äspö (Aberg)

At Äspö, SKB built a Hard Rock Laboratory (the Äspö HRL) between 1986 and 1995. A large quantity of data were collected before and during construction of the underground facility. The laboratory is used today for research purposes and data on the rock are gathered continuously from various research projects in the laboratory. Only data from the pre-investigations and from the construction phase are used in SR 97, however.

Äspö is an island in the archipelago situated approximately 2 km north of the Simpevarp Nuclear Power Plant in the municipality of Oskarshamn. The landscape is flat with a thin soil layer on highlands and bogs in lowlands. Most of the area is forested, but there is also cultivated land and pastureland. The elevation difference between the highest and lowest (sea level) point in the region around Äspö is about 30 m. The rate of land uplift is 1–2 mm per year.

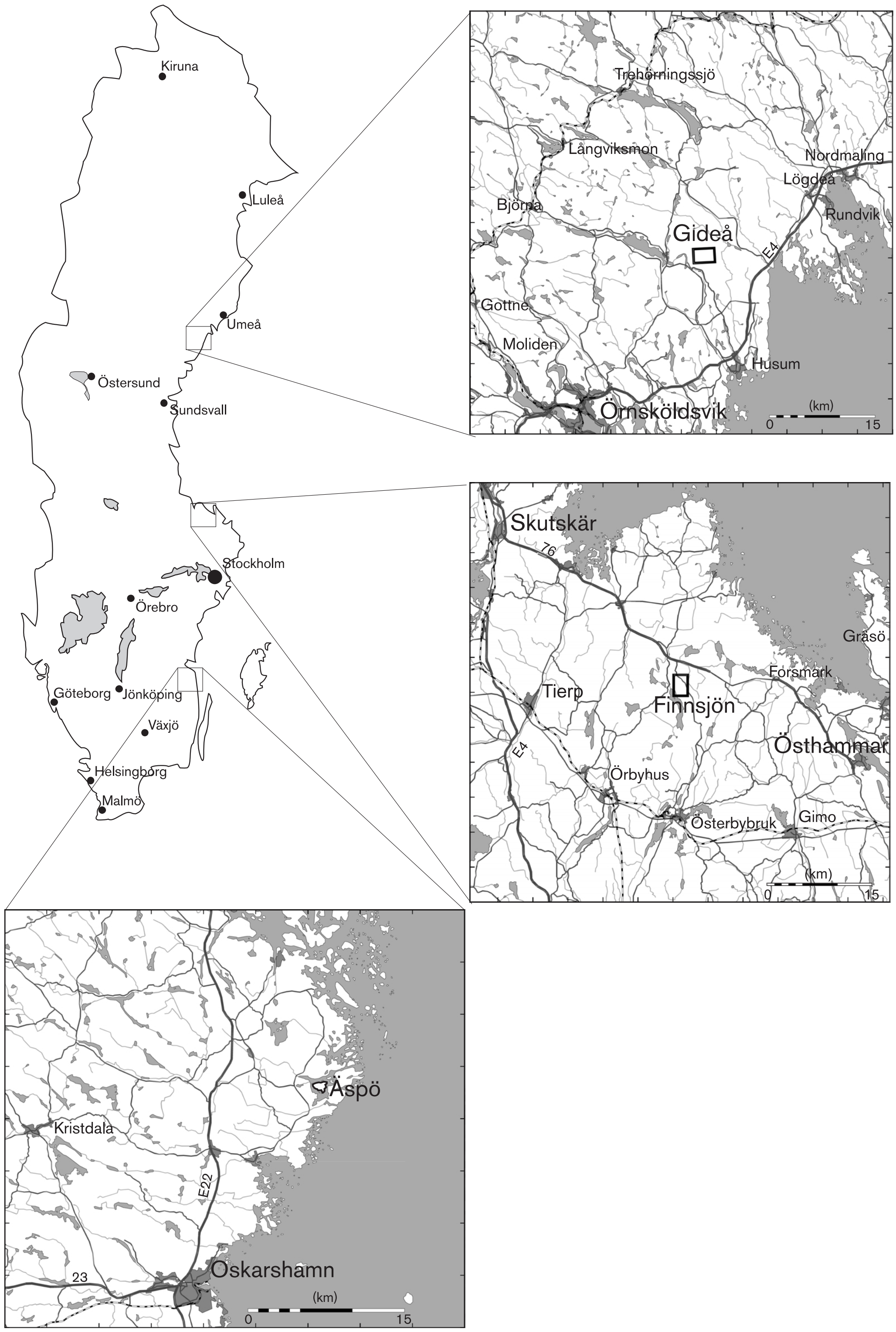

Figure 2-3. Data for the hypothetical repository sites Aberg, Beberg and Ceberg are taken from Äspö, Finnsjön and Gideå, respectively.

Approximately 1,800 million-year-old Småland granites dominate the bedrock in the region. Subordinate rock types are gabbro and diorite, as well as clasts of older rock types. A so-called younger granite, with an age of 1,400 million years, exists in the form of several massifs north and south of Äspö. There are four rock types on the actual island of Äspö: Äspö diorite, Ävrö granite, greenstone and fine-grained granite. The first two are the most common and are interpreted as variants of Småland granite.

The region contains kilometre-wide height area bordered by large valleys, which are normally interpreted as fracture zones. The frequency of regional fracture zones in and around Äspö is high compared with the surrounding region. There are few observed regional plastic shear zones. An exception is an approximately 100 m wide northeast-trending zone that runs through the central portion of Äspö.

The frequency of interpreted and observed regional and local fracture zones within Äspö is relative high.

The transmissivity of the fracture zones, as measured in boreholes on Äspö, varies between approximately 10^{-4} and 10^{-7} m^2/s. The hydraulic conductivity in the rock mass between the fracture zones on Äspö is on average around 10^{-8} m/s below 100 m. No change with greater depth has been noted.

The groundwater under Äspö is non-saline down to a depth of approximately 200 metres, below which the salinity increases and is about 11,000 mg/l at a depth of 500 metres, which is twice as high as today's Baltic Sea water.

Finnsjön (Beberg)

Finnsjön was investigated mainly during the period 1977–1978. The area was one of three that were studied to obtain data for the KBS-3 projects. Most of the knowledge that exists today from the area stems from a research project that was conducted during the years 1985–1989 in the northern part of the area. The purpose of the project was to study the geological and hydrogeological character of a flat permeable fracture zone and evaluate its importance for a repository for spent nuclear fuel.

Finnsjön is situated just north of Österbybruk, in the municipality of Tierp. The distance to the Baltic Sea is about 14 km. The landscape in and around Finnsjön is flat with rock outcrops, mires and small lakes. There are small cultivated areas in its environs. The rate of land uplift is 5–6 mm /y.

The oldest rock types in the region are approximately 1,900 million-year-old altered sedimentary and volcanic surface rock types. The latter contain iron minerals. Most of the region consists of slightly younger plutonic rocks: gabbro, diorite, tonalite, granodiorite and granite. They have intruded into the older surface bedrock. All of the aforementioned rock types were affected by a deformation phase 1,850–1,780 million years ago. The bedrock in the Finnsjön area is dominated by granodiorite, which is normally grey with a northwesterly and steeply-dipping foliation. In tectonically affected portions the granodiorite is reddish.

The deformation phase gave rise to an extensive system of plastic regional shear zones, mainly trending northwest. This direction is also common for regional fracture zones, which, in combination with crossing zones, give rise to a block-like network of fracture zones in the region.

Finnsjön can be divided into a northern and a southern block, separated by a steeply-dipping fracture zone. The blocks are bounded on the west and east by wide regional zones. There is a gently-dipping fracture zone in the northern block with a thickness of about 100 metres whose top surface is located at a depth of 100–300 metres. The zone has not been encountered in boreholes in the southern block. Several smaller fracture zones occur within both rock blocks.

As far as the transmissivity of the fracture zones is concerned, the gently-dipping zone stands out with a transmissivity of approximately 10^{-3} m^2/s. The equivalent value for other fracture zones varies between approximately 10^{-4} and 10^{-8} m^2/s. The hydraulic conductivity in the rock mass between the fracture zones on levels below a depth of 100 metres averages about 10^{-8} m/s. Flow measurements and other observations indicate a high flow in the upper portion of the gently-dipping zone. Calculations based on these measurements indicate a natural flow in this part of the zone of between 5 and 10 litres/second over a cross-section of 1,000 metres. Below the upper part of the zone, similar measurements reveal a very low natural flow.

In the northern block, the groundwater is non-saline down to the top surface of the gently-dipping zone. Below that the groundwater is saline, approximately 9,000 mg/l, which is twice as much as today's Baltic Sea water. Based on carbon-14 and tritium data, all samples from the saline water exhibit a very long residence time and a low fraction of meteoric groundwater. In the southern block, where the gently-dipping zone is missing, the groundwater is non-saline down to the end of the boreholes (about 700 m). Here modern hydrogeochemical analyses are lacking.

Gideå (Ceberg)

Gideå was investigated during the period 1981–1983. The area was one of four that was studied to obtain data for the KBS-3 projects.

The topographical relief in the region around Gideå varies from sea level to approximately 300 metres, making it much greater than for Äspö and Finnsjön. The Gideå area is located approximately 100 metres above sea level and within a large highland plateau. The distance to the sea is approximately 10 km. The area is dominated by forest, bogs and mires. The rate of land uplift is about 8 mm/y.

Mineralogically, the region is dominated by sedimentary gneiss (metagreywacke) formed about 1,950–1,870 million years ago. Older and younger granite is also present in the form of large massifs in the region. The youngest rock type is sills and steeply-dipping dykes of dolerite aged 1,270–1,214 million years. Sedimentary gneiss and regimes with older granite dominate within the Gideå area. They are intersected by a system of east-westerly, steeply-dipping dolerite dykes that can be up to 15 metres wide. The direction of foliation in the gneiss varies, but is mostly east-west with a medium-steep dip.

The frequency of interpreted and observed fracture zones within Gideå is relatively moderate.

The transmissivity of the fracture zones is lower than on the two other sites, varying between approximately 10^{-5} m^2/s and 10^{-8} m^2/s. The hydraulic conductivity in the rock mass between the fracture zones is also relatively low, averaging around 10^{-10} m/s at levels below 100 metres deep.

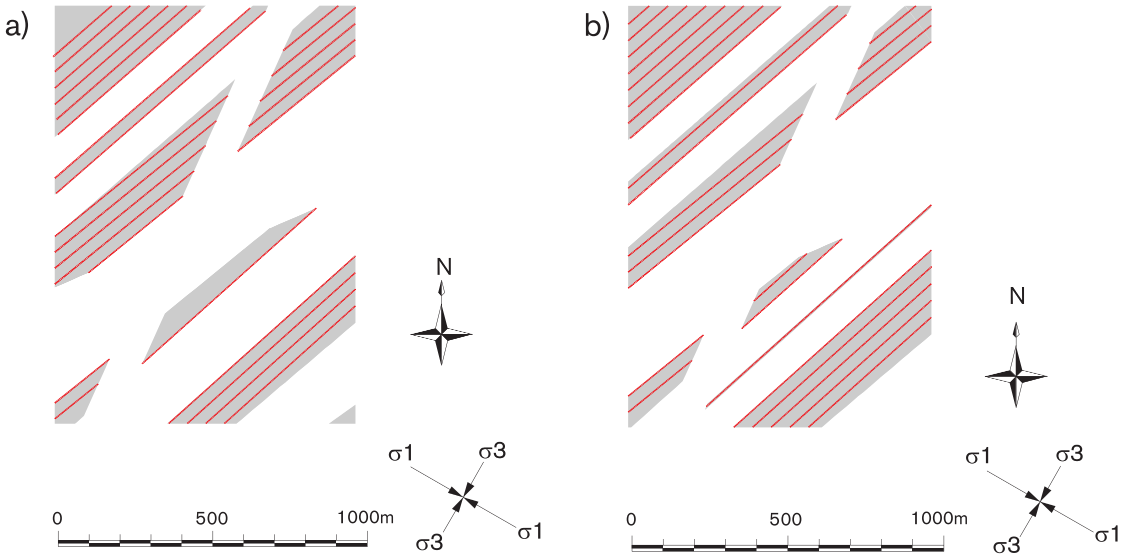

***Figure 2-4.** Repository layout for Aberg. The figure shows cross-sections at depths of a) 500 m and b) 600 m.*

Gideå is situated below the highest coastline, which means that the area was covered by water after the most recent ice age. The highest parts of the area rose out of the sea about 8,000–6,000 years ago, at which time the sea was non-saline or weakly saline. Two sampled sections also contain weakly saline groundwater. Other sections have non-saline water.

2.5.3 Site-adapted repository layouts

The layout of rock caverns, tunnels and deposition positions in the repository system is based on principles first presented in the KBS-3 study. A number of possible repository locations have been proposed for Aberg, Beberg and Ceberg, and the main alternative for each site is presented below.

In Aberg, the proposed repository layout is split into two levels at depths of 500 and 600 metres so that the entire repository can be fit into the relatively limited study site, see Figure 2-4.

In Beberg, the proposed repository is located 600 m below sea level so as to avoid a dominant horizontal structure on the site with good margin, see Figure 2-5. The deposition tunnels are oriented perpendicular to the maximum horizontal stress. This direction has been chosen to avoid long intersections with water-bearing fractures with the same direction as the horizontal stress. This layout resembles the one used as an example in the SKB 91 safety assessment.

In Ceberg, the repository located at 500 m below sea level, i.e. around 600 m below the ground surface, see Figure 2-6. The deposition tunnels are oriented perpendicular to the maximum horizontal stress.

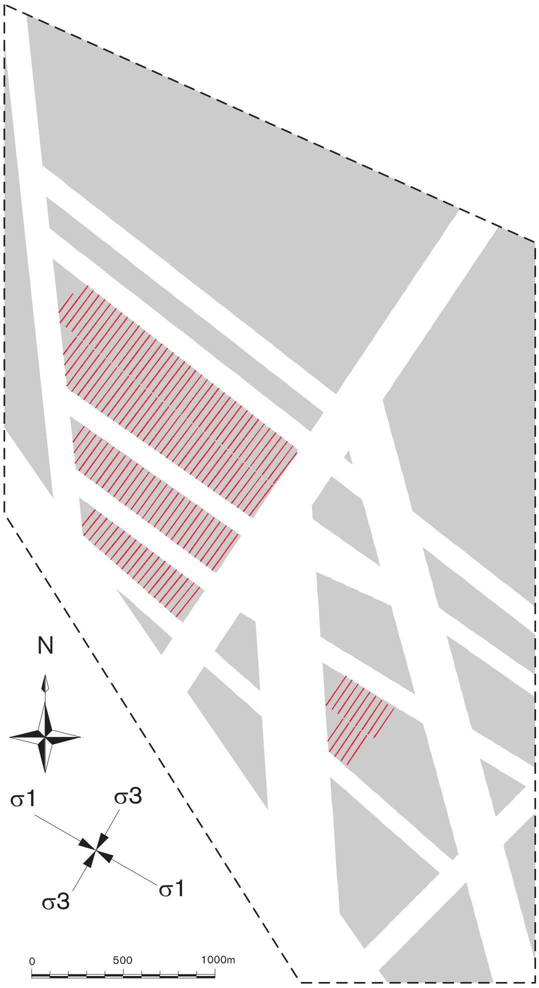

__Figure 2-5.__ Repository layout for Beberg with deposition tunnels oriented perpendicular to the maximum horizontal stress. The depth is 600 m.

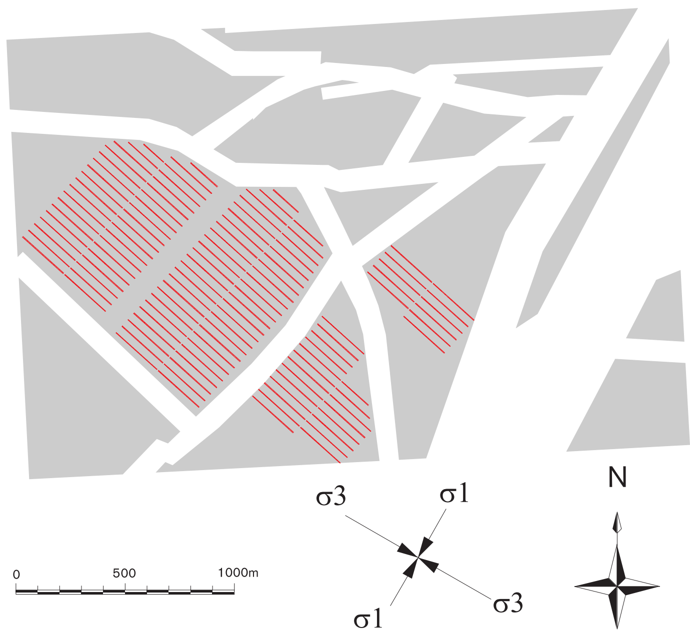

__Figure 2-6.__ Repository layout for Ceberg with deposition tunnels oriented perpendicular to the maximum horizontal stress. The depth is 500 m.

2.5.4 THMC description of processes and variables

In the complete THMC description of the geosphere, important site-dependent variables are e.g. fracture structure, groundwater composition and rock stresses. Important geosphere-related processes are groundwater flow, the mechanical processes that determine the rock's long-term stability, and the chemical transport and reaction processes that determine the groundwater composition in the long term and thereby the chemical environment for buffer and canister.

2.6 Safety criteria

According to section 1.8.1, the safety assessment describes what changes the repository system undergoes with time and what the safety-related consequences of this are. The principal safety functions of a deep repository concern its isolating and retarding capacities. The acceptance criteria say that the risk of deleterious effects may not exceed 10^{-6} per year, see section 1.5.1.

An overall goal of the safety assessment is therefore to determine whether this criterion has been satisfied. The criterion is general and not tied to any special repository design. Based on the design of the KBS-3 system and its intended long-term functions, more specific "auxiliary criteria" can be formulated for long-term safety. These auxiliary criteria do not determine safety directly, but serve as "guideposts along the road" in the actual safety assessment.

Canister

The first safety criterion is that the canister must remain intact for a very long time, i.e. its copper shell must not be breached.

Another criterion is that the temperature on the canister surface may not exceed 100°C to avoid boiling on the canister surface. Boiling could lead to enrichment of salts on the surface, which could in turn cause corrosion effects that are difficult to analyze.

There are also **design requirements** on the canister, which are made to ensure that the isolation function will last for a long time. The design requirements are of assistance in selecting the canister's materials and dimensions, but are not used directly in the safety assessment.

Buffer

For the buffer as well, there are various requirements which are used to determine suitable materials and dimensions. Few of these requirements are used in the actual safety assessment. Instead, the chosen buffer is evaluated under the different conditions that will prevail in the scenarios.

The overall purpose of the buffer is to serve as a diffusion barrier between canister and rock. To fulfil this purpose, the buffer must have a low hydraulic conductivity (water-conducting capacity) and must be able to keep the canister centred in the deposition hole for a long time. It must also "self-heal" minor cracks in the clay that may arise e.g. initially during the saturation process.

Backfill

The long-term safety criteria for the backfill say that it should have a hydraulic conductivity that does not significantly exceed the average conductivity of the surrounding rock, and a swelling pressure of at least 0.1 MPa against the tunnel roof to support the rock around the tunnels.

Geosphere

A criterion for the geosphere is that the groundwater at repository depth must be oxygen-free for a very long time. This is required to prevent oxygen corrosion of the copper shell, but also because it generally simplifies the analysis of the evolution of chemical conditions.

Otherwise it is difficult to formulate criteria for the geosphere that can be used directly in the safety assessment. The geosphere does indeed contribute to various safety functions: Its mechanical stability is important for isolation, the groundwater composition is of importance for corrosion processes and thereby isolation, etc. The problem as far as criteria are concerned is that many different measurable properties combine to contribute in a complex fashion to the functions of the geosphere. Criteria of the kinds formulated above for canister and buffer are therefore not used for the geosphere when the long-term evolution of the repository is studied in the safety assessment.

It can, however, be meaningful in site investigations to formulate both requirements and preferences regarding various aspects of the geosphere's initial state, i.e. conditions which can be directly or indirectly observed in investigations of candidate repository sites.

Safety criteria and variables

The safety criteria put limitations on a number of variables (geometry of the copper shell, temperature on the canister surface, swelling pressure and smectite content of the buffer, oxygen content of the groundwater) that should not be exceeded in the long term when the evolution of the repository is analyzed. Smectite content and swelling pressure will, for example, be affected by the fact that the evolution of chemical conditions leads to material changes in the buffer, the canister surface is heated due to the decay heat in the fuel, etc.

The overall criterion is, however, always the limits on releases from the repository formulated by the safety and radiation protection authorities.

2.7 Completeness of system description

An important question for the safety assessment is whether all relevant processes have been identified in the system description. It can never be strictly proved that the system description is complete, i.e. that it covers all processes and interrelationships relevant to the evolution of the repository. Confidence in the completeness of the system description must instead be subject to assessment.

Such an assessment can be based on both general arguments regarding the maturity of the science underlying the system description and specific arguments regarding the structure of the system description in a safety assessment.

Much of the understanding of the system rests on scientific foundations which in many cases have remained unchanged for a long time. This is true, for example, of knowledge in the fields of radioactivity, thermodynamics and hydrodynamics (groundwater flow). Additional knowledge comes from a couple of decades of research and development on technical and scientific questions related specifically to the deep repository. The international exchange of experience is an important part of the accumulation of knowledge and understanding of the evolution of the repository system.

Concrete administrative measures that can be adopted to achieve completeness in a system description are:

- Database management and other systematic documentation of all processes that have been identified as being important for the evolution of the repository.
- Comparisons with databases established by other organizations and in joint international projects.

In addition to a well-developed scientific knowledge base and advanced database management of relevant information, the experts who carry out system description and safety assessment must also be highly qualified. Competency is required in both methodology and understanding of the function of the repository system.

Even though completeness cannot be proved, the consequences of an incomplete description can to some extent be illustrated. Assume, for example, that a process which rapidly degrades the copper canister was unidentified. The consequences of this can be illustrated by performing a calculation of radionuclide transport where the canister's isolation is assumed to be completely lost say 1,000 years after closure. Such calculation cases are reported in the canister defect scenario.

The structure of the system description in SR 97, with choice of processes etc., was described briefly in section 2.1.1. It is SKB's judgement that all important processes in the evolution of the repository system are represented in the system description.

3 Scenarios

3.1 Choice of scenarios

With the system of processes defined in the system description, the repository's evolutionary pathway is determined if:

a) an initial state and

b) the long-term conditions in the surroundings

can be determined. Both initial state and conditions in the surroundings are, however, associated with uncertainties.

Current external conditions can be observed and described. This includes, for example, what the biosphere above the repository looks like, the climate, the situation in distant parts of the geosphere and the structure of society. In contrast, great uncertainty exists with regard to how climate, biosphere and above all society will change in the future. The consequences of such changes for the repository must nevertheless be analyzed. The method used for this is to analyze the evolution of the repository for a number of possible future situations in the surroundings. A set of different scenarios is analyzed.

In SR 97, five scenarios are analyzed. The selection is based on experience from the work with so-called interaction matrices, on information in databases regarding factors and conditions that are relevant to safety, and on the system description. Experience from previous safety assessments by SKB and other organizations has also been drawn on. The choice is an expert assessment of which initial conditions and conditions in the surroundings are important to analyze in a safety assessment.

The scenarios chosen for SR 97 are:

- A base scenario where the repository is conceived to be built according to specifications, where no canisters have initial defects and where present-day conditions in the surroundings are assumed to exist.
- A canister defect scenario which differs from the base scenario in that a few canisters are assumed to have initial defects.
- A climate scenario that deals with future climate-induced changes.
- An earthquake scenario.
- A scenario that deals with future human actions that could conceivably affect the deep repository.

The choice of scenarios in SR 97 is based on the system description and experience from previous work. Figure 3-1 illustrates the different scenarios schematically.

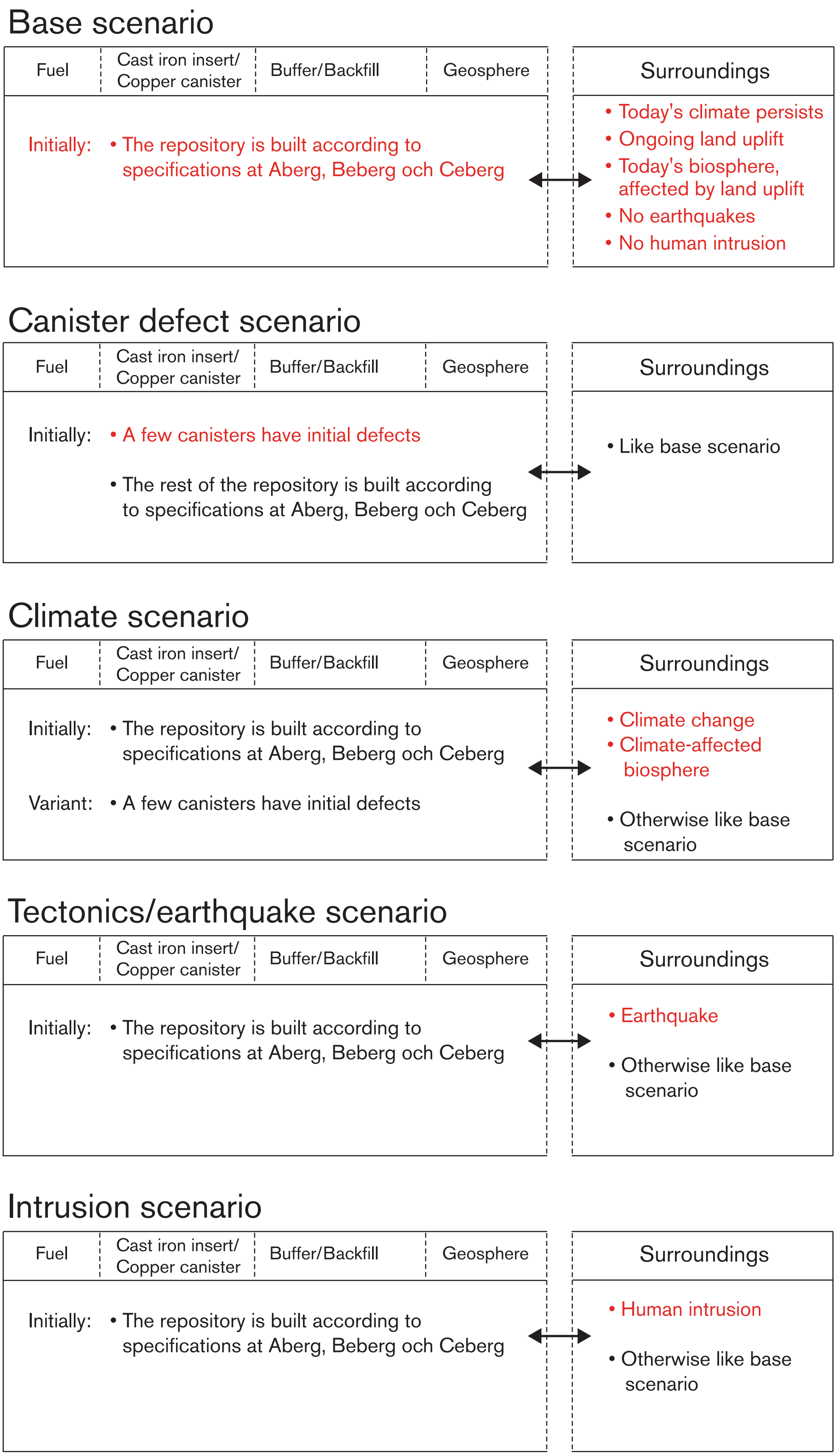

Figure 3-1. *The different scenarios in SR 97.*

In SR 97, the base scenario is a ground for comparison for other scenarios. Today's climate is assumed to prevail, even though it is known that the climate is undergoing long-term change. Another possibility would be to include future climate changes in the base scenario in order to achieve a more realistic ground for comparison. This has not been done in SR 97 for the following reasons:

- Scientists and experts do not agree today on what climate changes can be expected. This goes beyond the debated greenhouse effect and includes uncertainty in predictions of the timing and above all the extent of future periods of colder climate.
- It is relatively simple to describe the influence of today's climate on the repository system. This puts the focus of the base scenario on the internal evolution of the repository. This results in a gradual build-up of the complexity of the analysis of repository evolution: First the evolution is analyzed with simple boundary conditions in the base scenario, and then for more complex conditions in the climate scenario. This should both be pedagogically enlightening and permit a clearer assessment of which internal and external conditions might disturb repository function.
- SSI's regulations state explicitly that the safety account shall include a case where the biosphere conditions do not change.

On the other hand, there is in principle nothing to prevent the inclusion of climate change (or other changes in conditions in the surroundings) in the base scenario.

3.1.1 Completeness/coverage in choice of scenarios

Just as it is not possible to prove that the system description is complete, it is impossible to prove that all states in the surroundings that could affect the repository have been identified. Confidence must instead be established in the fact that the scenarios together provide sufficient coverage for a fair assessment of the safety of the repository.

Important measures for achieving completeness in the choice of scenarios are to:

- ask the question for each process in the system description whether it can be influenced by the initial state or the external surroundings,
- judge for each variable in the system description whether uncertainties in the initial value warrant the choice of specific scenarios or variants,
- systematically document the conditions in the surroundings that have been identified over time as being important for repository evolution,
- compare with scenarios and databases established by other organizations and in international cooperation.

The choice of scenarios in SR 97 is based on all these principles. The system description and databases and experience from previous safety assessments by SKB and other organizations have been utilized.

Uncertainties in the initial state have been handled in the following manner in the choice of scenarios:

- **Fuel:** No uncertainties in the initial state have been judged to be so essential for the outcome that they should be handled in a scenario of their own. Uncertainties in e.g. radionuclide inventory and decay heat can, if necessary, be analyzed as variants of the base scenario (thermal evolution) and the canister defect scenario (radionuclide transport).

- **Canister:** The uncertainties pertain above all to initial integrity. This is handled by analyzing the effects of initial damage in a separate scenario.
- **Buffer and backfill:** No uncertainties in the initial state have been judged to be so essential that they are handled in scenarios of their own. SR 97 does not cover analysis of accidents in fabrication or deposition of buffer. It should be possible to avoid such "buffer defects" by means of suitable inspection procedures in connection with fabrication or deposition of the buffer. Furthermore, the effects are judged to be much less serious than those of initial canister damage. If it should nevertheless be decided in future assessments to study the consequences of "buffer defects", this can suitably be done as a variation within the base scenario.
- **Backfill:** The backfill material must be adapted to the hydrogeochemical conditions on the repository site, above all the salinity. No such site-specific adaptation, and therefore no site-specific description of the initial state of the backfill, has been done in SR 97. The analyses of post-closure evolution and performance of the backfill are sometimes simplified. Uncertainties in the initial state of the backfill can, if necessary, be analyzed as variations within the base scenario in future safety assessments.
- **Geosphere:** There are great uncertainties regarding the fracture structure and the spatial distribution of hydraulic properties in both fracture zones and the rock mass in the initial state. The uncertainties are above all of importance to the analysis of radionuclide transport in the canister defect scenario. The uncertainties are handled by studying a number of variants in the detailed hydraulic analyses of the geosphere in the canister defect scenario.

SKB believes that the chosen scenarios provide good coverage of future evolutionary pathways for the deep repository. Nevertheless, the choice of scenarios and variants in future assessment may be modified as more knowledge and experience are acquired.

3.2 Base scenario

The base scenario describes the expected course of events (evolution) for the case where the repository is built according to specifications and the conditions in the surroundings are assumed to be in principle constant and the same as today's. All canisters are assumed to be without fabrication defects, and today's climate persists in the future as well.

Based on previous safety assessments, the canisters can be expected to last for a very long time given these premises. Radionuclide transport should therefore not have to be dealt with in the base scenario. However, this is not a premise, but must be demonstrated by means of the analyses performed for the scenario.

The overall purpose in the base scenario is to study the isolating function of the canister. A fundamental requirement on the repository system is that the canisters' copper shells shall remain unbreached. If this requirement is met, it alone is sufficient to demonstrate safety. To facilitate the analysis, other requirements are also made (see section 2.6).

3.2.1 Initial state and boundary conditions

For the initial state, the repository is assumed to be built according to specification and all deposited canisters are postulated to be without fabrication defects. In summary, the following is assumed in the base scenario regarding the repository's surroundings:

- Present-day climatic conditions are assumed to prevail in the future.
- Land uplift and its influence on the biosphere in particular is included in the base scenario. Influence on groundwater flow is discussed summarily.
- Present-day site-specific biospheres are assumed to persist, except for the effects of land uplift on the biosphere.
- Rock-mechanical changes take place only as a result of aseismic processes, i.e. earthquakes are not included in the base scenario.
- No human intrusions occur.

3.2.2 Overview of processes and dependencies

The initial state of the repository is changed with time by a number of processes that take place in fuel, canister, buffer/backfill and geosphere. The processes can be divided into the categories radiation-related (R), thermal (T), hydraulic (H), mechanical (M) and chemical (C). Figure 3-2 shows a simplified diagram of the system of processes.

Many processes proceed in a parallel and coupled fashion, making the situation appear very complex. However, a closer analysis of the essential couplings in the system clearly reveals main features which greatly simplify the treatment in the safety assessment.

The radiation-related processes – radioactive decay and radiation attenuation – determine what the radiation intensity is in different parts of the repository. They are largely independent of all other processes and can therefore be described first.

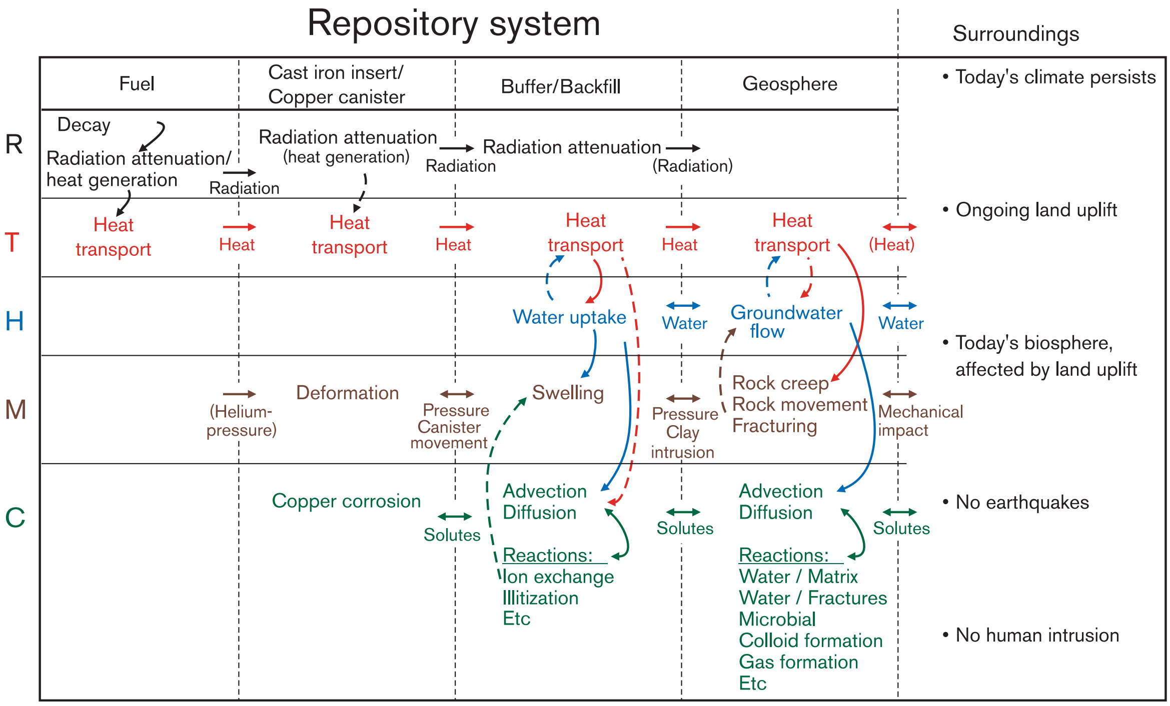

Figure 3-2. Main features of the process system for the base scenario.

After the radiation-related processes have been quantified, the thermal evolution of the repository can be determined in its essential respects. Radioactive decay constitutes the source of the heating, and the further thermal evolution is controlled by heat transport in the different parts of the repository. Heat transport is dependent on the material properties of the repository parts, which are largely constant over long time spans with one important exception: The thermal properties of the buffer are dependent on its water content, which changes during water saturation of the buffer.

The hydraulic evolution in the base scenario concerns only the buffer and the geosphere, since the interior of the canister and the fuel are hydraulically isolated by the copper shell. The original flow conditions in the geosphere are expected to be restored within a period of between ten and one hundred years after repository closure. The process of water saturation of the buffer/backfill is dependent on the influx of groundwater to individual deposition holes and tunnels. Thus, in the case of the buffer there is an interaction with the thermal evolution.

The mechanical evolution of the repository can then be determined. The fuel is mechanically isolated from its surroundings by the cast iron insert. The mechanical evolution is dominated to begin with by swelling of the buffer, which is determined by the buffer's water content, and the thermal expansion of the rock, which is determined by the temperature change in the rock. Swelling of the buffer gives rise to mechanical loading of both canister and rock. In the long term, the mechanical evolution is controlled by large-scale changes in the geosphere.

Finally, the chemical evolution of the repository can be described. The fuel and the interior of the canister are isolated from their surroundings by the copper shell in chemical respects as well. The chemical evolution is dominated by reaction and transport processes in buffer and geosphere and by corrosion of the outside of the copper canister.

The radiation-related, thermal, hydraulic, mechanical and chemical evolution of the repository are described in that order in the following sections.

3.2.3 Radiation-related evolution

The radiation-related evolution includes radioactive decay in the fuel, the radiation to which this gives rise, and the spreading and attenuation of the radiation in fuel, canister and buffer.

The following aspects are described in the base scenario:

- The fuel's content of radionuclides and its radiotoxicity as a function of time.
- The heat output of the fuel as a result of radioactive decay.
- The levels of γ and neutron radiation in fuel, canister and buffer.

Importance for safety: The radiation-related evolution of the repository does not include any processes that have a direct bearing on the isolating capacity of the repository. The radiation does not affect the material properties of the canister or the buffer in the long term. The results are used as input data to other parts of the analysis. The decay heat comprises the source of the thermal evolution, and the radiation can give rise to radiolysis effects in the buffer's pore water in the base scenario.

As an example of results, Figure 3-3 shows how radiotoxicity changes with time in the spent fuel.

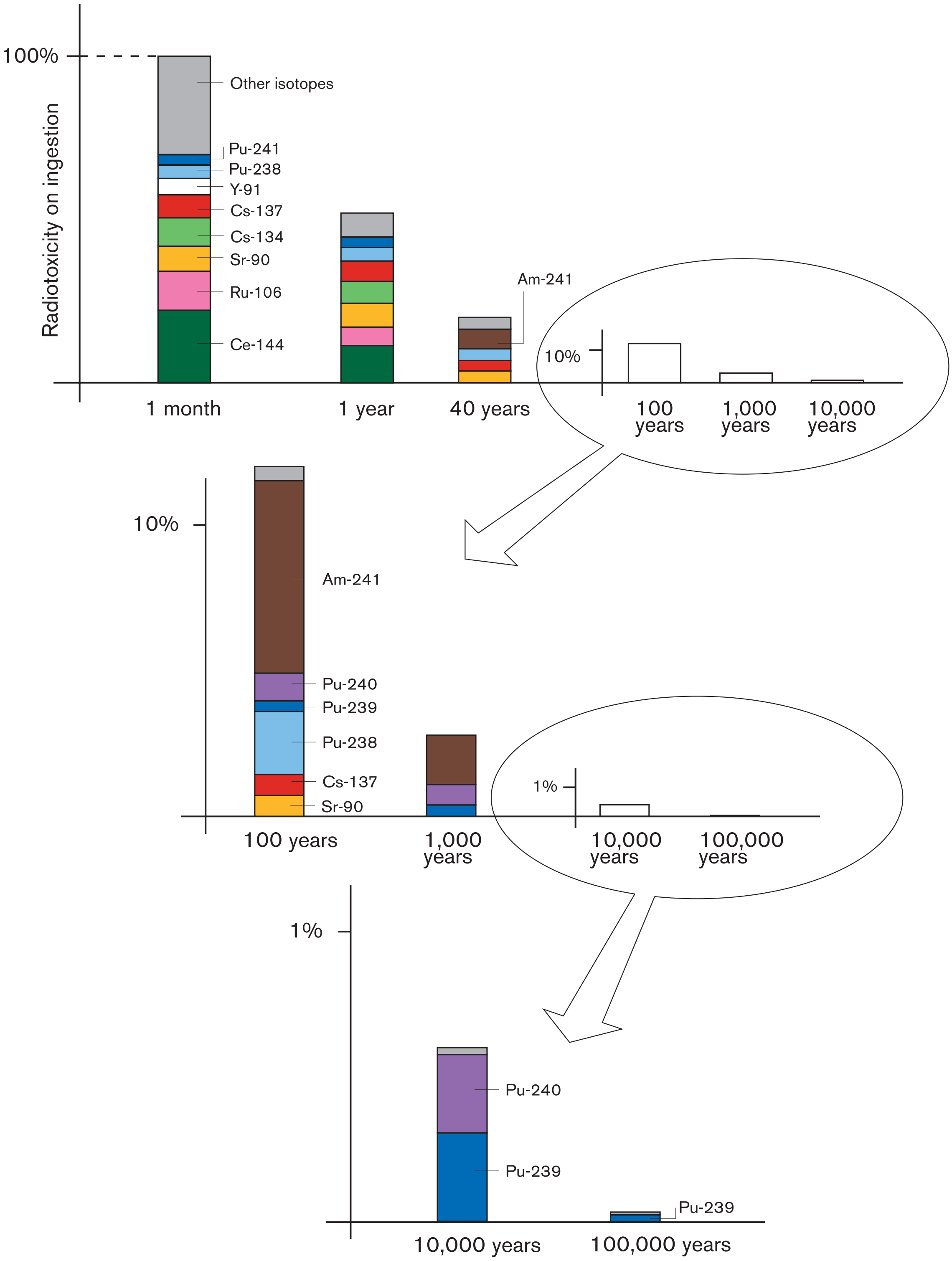

Figure 3-3. Relative radiotoxicity on ingestion of BWR fuel with a burnup of 38 MWd/kg uranium.

Confidence

Process understanding: The fundamental processes in the radiation-related evolution of the repository – radioactive decay and radiation attenuation – are well-understood both experimentally and theoretically, see the Process Report.

Models: The evolution of radioactivity, radiotoxicity and decay heat can be calculated analytically when the content of radionuclides at deposition is known. Models for radiation shielding calculation provide a sufficiently detailed treatment of radiation attenuation.

Data: The radiation-related evolution requires decay data and inventories of radionuclides in the fuel, as well as radiation shielding data for the fuel, canister and buffer materials. The quality of available data is fully adequate for the calculations carried out in the base scenario. Uncertainties in the radionuclide inventory are discussed in the Data Report.

Conclusions

The radiation-related evolution has no direct bearing on safety in the base scenario. The calculation results are used in other analyses, particularly in the analysis of the thermal evolution, where the decay heat is used.

Our fundamental understanding of the processes involved is good, as is the quality of available models and data.

3.2.4 Thermal evolution

The thermal evolution describes how heat is transported from the fuel out into the other parts of the repository. The following aspects are described in the base scenario:

- An analysis of the temperature on the outside of the canister, which may not exceed 100°C. The requirement is set to avoid boiling on the canister surface. Boiling could lead to enrichment of salts on the surface, which could in turn cause corrosion effects that are difficult to analyze.
- The temperature in the entire repository system as a function of time, to be used in the mechanical and chemical analyses in particular. The temperature impact at the ground surface above the repository also needs to be estimated.

Importance for safety: The thermal evolution of the repository does not include any processes that have a direct bearing on the isolating capacity of the repository. The results are used in the mechanical and chemical analyses.

Results

The evolution of temperature is calculated based on the decay heat in the fuel, and thermal and geometric data for all parts of the repository. It is pessimistically assumed that the buffer has not yet become saturated by water uptake from the surrounding rock, and that there is an air-filled gap between canister and buffer.

The results show that the canister reaches its maximum temperature around 10 years after deposition. The surface temperature of the canister always lies below 90°C. The result is expected, since the repository layout is chosen with the aid of thermal

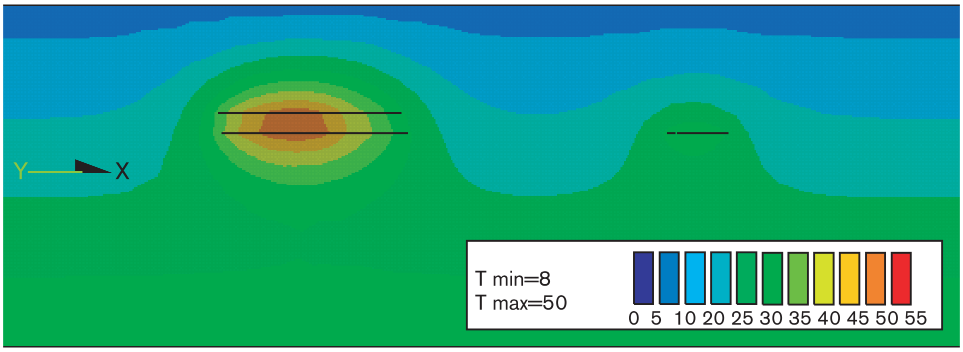

Figure 3-4. *The temperature in a vertical section at Aberg 1,000 years after the start of deposition. The horizontal black lines mark the repository. The top of the figure corresponds to the ground surface.*

calculations to ensure just this. The safety margin of 10°C to the limit of 100°C is set to allow for uncertainties in input data.

The temperature in the geosphere is calculated site-specifically as a function of time and space. The temperature at the wall of the deposition hole reaches a maximum value after about 20 years, at which point it is approximately 60, 50 and 45°C in Aberg, Beberg and Ceberg, respectively. Viewed over the entire rock volume of the repository, the maximum temperature is reached when heat waves from different tunnels interfere, and in Aberg when heat from the two repository levels is superimposed. The maximum temperature in the rock blocks between the repository levels in Aberg is about 55°C and is reached after 450 years. The corresponding temperatures in the repository levels in Beberg and Ceberg are 45 and 40°C, respectively, and are reached after 90 and 80 years, respectively. Figure 3-4 shows the temperature in a vertical section at Aberg after 1,000 years.

Heat from the repository reaches the surface after a few hundred years. The heat will have a marginal impact on the thermal conditions on the ground surface. The effect is comparable to the natural geothermal heat flow, which is in turn less than a tenth of a percent of the heat input of sunshine.

Confidence

Process understanding: Our fundamental understanding of different heat transport phenomena is good for canister, buffer and rock, see the Process Report.

Models: Available models adequately represent heat transport in the different parts of the repository. There are adequate both analytical and numerical models for the geosphere.

Data: The analysis requires data on the decay heat from the fuel (from the radiation-related evolution) as well as thermal and geometric data for canister and buffer. In addition, site-specific thermal data for the geosphere are required. In general, data on heat transfer between different media, e.g. cast iron insert and copper canister or canister and buffer, are more uncertain than heat conduction data within a medium.

Data uncertainties are handled pessimistically in the estimation of the canister's surface temperature. Otherwise the quality of the data is adequate for the calculations required in the base scenario.

Conclusions

The requirement that the surface temperature of the canister may not exceed 100ºC can always be met with the necessary safety margin by choosing a suitable spacing between deposition holes or by adjusting the fuel content of the canisters. The requirement is verified by means of standardized temperature calculations where data uncertainties are handled pessimistically.

Calculations of other aspects of the repository's thermal evolution can be conducted with sufficient precision. The results have no direct bearing on safety, but are used above all in the description of the mechanical and the chemical evolution.

3.2.5 Hydraulic evolution

The hydraulic evolution in the base scenario concerns only buffer/backfill and geosphere as long as the canister is intact. The canister interior and the fuel are hydraulically isolated by the intact canister.

The hydraulic evolution in the base scenario describes how water and gas are transported in the fracture system in the geosphere and in the pore system in the buffer/backfill.

The following are needed in the base scenario:

- a description of the site-specific hydraulic evolution in the geosphere, and
- a more detailed description of the hydromechanical evolution when buffer/backfill is saturated with water.

Importance for safety: The hydraulic evolution in the geosphere has an indirect bearing on the canister's isolating capacity, since the groundwater transports solutes and thereby affects chemical processes such as canister corrosion.

The hydraulic evolution of the buffer does not have a direct bearing on the canister's isolating capacity. However, it is important to ensure that the buffer is saturated under any circumstances so that it will function as intended in the repository. Furthermore, it is important to study the build-up of the swelling pressure as the buffer is saturated, among other things so that it is possible to determine what stresses an uneven swelling pressure might exert on the canister.

Groundwater flow

Initially the geosphere around the repository is partially drained as a consequence of construction. Reversion to the original groundwater levels and flow patterns proceeds as different parts of the repository are built, backfilled and closed after deposition.

In the geosphere, a more or less constant flow state is reached after the groundwater has returned to its original level.

Model studies of the undisturbed, natural regional groundwater flow at the three repository sites are reported in the base scenario. Flow patterns, dominant flow paths, flows at repository depth, the importance of density variations caused by varying salinity and other factors are discussed here. More detailed, local flow studies are reported in the canister defect scenario.

In addition, the expected long-term evolution on the three sites is discussed. The changes will be greatest on the island of Aberg, which is projected to become a part of the mainland in about 2,000 years as a result of land uplift. A model study predicts slightly increased flows and a shift towards less saline water in a ten-thousand year perspective.

Process understanding is good and the quality of models and data for the hydraulic evolution in the base scenario is judged to be adequate. A more detailed discussion can be found in section 3.3.5 on the canister defect scenario.

Hydromechanical evolution in buffer/backfill

The buffer and backfill are partially saturated with water at deposition. After deposition, they will eventually become saturated with water from the surrounding rock. The saturation process in the buffer, with its coupling to the thermal (heat flow from the canister) and mechanical (swelling pressure) evolutions, is studied in an integrated modelling.

A typical case is reported, along with several variants with, among other things, different assumptions regarding the hydraulic properties of the geosphere, and whether water-bearing fractures intersect the deposition hole. Figure 3-5 shows the degree of saturation at different times for the typical case. Full water saturation is reached here after about 12 years.

Pressure build-up and movements of the canister and the buffer are also calculated in the model. The calculations show that the canister moves upward approximately one centimetre in the buffer and returns to its original position after about 10 years. The buffer intrudes permanently about 8 centimetres into the backfill in the deposition tunnels.

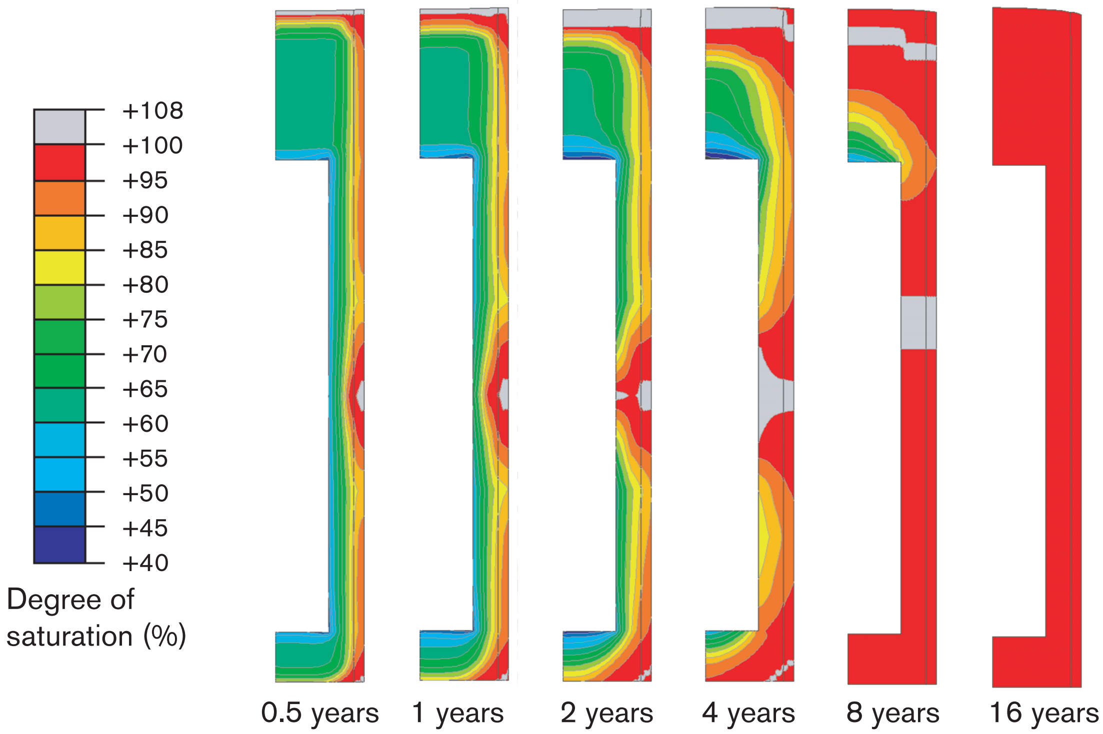

Figure 3-5. *Degree of saturation in the buffer at different times.*

The time required to achieve full water saturation lies between 6 and 35 years for all variations, except the one where the rock mass has extremely low conductivity. This difference is of no practical importance for the performance of the repository. In the case with extremely low conductivity, the buffer is expected to be saturated by water inflow from the backfill, but this process has not been modelled.

Exactly what happens in the gap between canister and buffer during the initial phase of saturation is uncertain. As a result, heat transfer in the gap must be handled pessimistically when the maximum temperature on the canister surface is calculated.

The differences in the hydraulic conductivity of the rock mass at Aberg, Beberg and Ceberg influence the water saturation time. It is likely that differences in hydraulic properties between different deposition holes on the same site are greater than the differences between the sites, since all sites are expected to have some deposition holes in very impervious rock without water-bearing fractures.

After water saturation the hydraulic conductivity of the buffer is very low, which contributes to its isolating function in the repository.

Confidence

Process understanding: Our understanding of the processes that drive water transport in unsaturated buffer and backfill material and our knowledge of how the processes are influenced by various factors is good enough to carry out reliable model calculations in the safety assessment, see the Process Report.

Models and data: The calculations require hydraulic and mechanical data on buffer/backfill and data on the hydraulic conditions in the rock around deposition holes, which constitutes the predominant uncertainty in the description of the hydraulic evolution of the repository.

The quality of input data and models used in the calculations is judged to be sufficient to carry out an adequate analysis of the hydraulic evolution in the base scenario. Precision in the calculations would increase with better knowledge of the hydraulic properties around individual deposition holes, which cannot be expected until the repository has been built. Long-term hydromechanical consequences of chemical changes in the buffer and backfill materials are discussed in section 3.2.7.

Conclusions

Full water saturation in the buffer is reached for a variety of hydraulic conditions in the near-field rock within ten years or so. When full water saturation has been reached, the buffer has the hydraulic and mechanical properties that are crucial to its long-term performance in the repository. Its hydraulic conductivity is less than 10^{-12} m/s, which means that diffusion is the dominant transport mechanism between canister and rock. The swelling pressure has built up to approximately 8 MPa, which means that the buffer can “self-heal” any damages.

Confidence in understanding, models and data is deemed to be sufficient for the base scenario.

3.2.6 Mechanical evolution

The mechanical evolution in the base scenario includes mechanical changes in the geosphere (stress changes, rock movements and possible fracturing), in both the short and long term and on different scales. The mechanical evolution in the buffer includes above all swelling and the movements in buffer/backfill caused by swelling. The mechanical evolution also includes an analysis of how the canister is affected by the loads to which it is subjected by buffer and rock.

The following are needed in the base scenario:

- an analysis of the long-term mechanical evolution of the repository rock
- an analysis of the development of the swelling pressure in buffer/backfill
- an analysis of how the canister is affected by the loads to which it is subjected by buffer and rock.

Importance for safety: The mechanical evolution may have a direct bearing on safety, since the mechanical stresses on the canister could affect its isolating capacity.

Geosphere

No site-specific rock-mechanical model studies have been conducted in SR 97. The analysis is instead based on previous studies and pessimistic arguments and approximations. Effects of earthquakes are not dealt with in the base scenario.

One of the most important purposes of the rock-mechanical analyses is to estimate rock movements at deposition holes. Rock movements take place primarily along existing fractures. An important premise for the analyses of movements around deposition holes is therefore that the positions of the deposition holes be chosen so that they are not intersected by large fractures or fracture zones. Based on the natural rock stresses on the three sites and the changes brought about by repository construction, the following is shown:

- The swelling pressure from buffer and backfill affect the stress levels marginally. On the other hand, the presence of buffer/backfill in the repository's cavities makes an important contribution to the stability in the rock around deposition holes and tunnels.
- According to earlier model calculations, thermal stresses that arise when the repository is heated by the decay heat in the fuel cause millimetre-sized fracture movements in the near field. The thermal stresses are greatest around 50 to 200 years after deposition. In major fracture zones, the movement may be a centimetre or so.
- In the long term, the mechanical load on the host rock is influenced by slow, large-scale movements in the bedrock as well as by the rock's own long-range material properties, which can lead to time-delayed deformations (creep movements). Large-scale movements can be roughly estimated to give average stress changes of 5 MPa in 100,000 years, which is small compared with the uncertainty inherent in the determination of the present-day natural rock stresses. The effects are judged to be much smaller than those of the thermal pulse. Long-term creep movements in the bedrock could above all lead to compression of deposition holes and tunnels. Not even hypothetical borderline cases of these effects are sufficient to jeopardize the mechanical stability of the repository.

Process understanding: Our fundamental understanding of the processes and conditions in the surroundings that control the mechanical evolution of the repository is not complete. This applies particularly to time-dependent deformations. However, our understanding is fully adequate for judging the mechanical evolution and stability in the base scenario if pessimistic approximations are employed.

Models: Serviceable numerical models are available for some processes which describe the mechanical behaviour of rock and rock masses and can be applied on both the near-field scale and the repository scale for the load cases that occur in the base scenario. There are as yet no fully applicable and validated models for time-dependent deformations and formation of new fractures. Rock-mechanical models are generally used to improve process understanding and to set bounds on the movements and deformations that may have a bearing on the safety of the repository, rather than to make accurate predictions.

Data: Accurate values exist for most rock-mechanical properties on the laboratory scale, but when the data are upscaled to large rock volumes the uncertainties can become considerable. Rock stress data from the field can also have considerable uncertainties. However, the data quality is fully adequate for the schematic and bounding type of analysis on which the conclusions in the base scenario are based.

Buffer/backfill

The swelling in the buffer was analyzed in conjunction with the hydraulic evolution of the buffer above. At full water saturation, the swelling pressure amounts to approximately 8 MPa.

Canister

The canister is subjected to mechanical loading via the buffer in the form of groundwater pressure and swelling pressure. If the pressure on the canister surface is evenly distributed, a BWR canister can take a load of 80 MPa and a PWR canister 110 MPa. This is far above the sum of swelling pressure (8 MPa) and groundwater pressure at a depth of 500 metres (5 MPa).

The swelling may also give rise to an unevenly distributed pressure. The strength of the canister has been analyzed for a number of hypothetical such cases during and after the buffer's saturation phase. The cases are pessimistic simplifications of not entirely unreasonable situations. None of the calculation cases for swelling in the buffer presented above give such high stresses as the cases for which the canister has been analyzed, however. The stresses were estimated by means of approximation or modelling. None of the calculation cases was found to cause damage to the canister.

In order to be able to judge the consequences of rock movements around deposition holes, the strength of the canister has also been calculated for a postulated displacement of 0.1 m lasting 30 days along a horizontal fracture. The results showed that rock movements of on the order of 0.1 metre do not lead to immediate canister failure. With pessimistic assumptions, however, the possibility cannot be ruled out that creep deformation in the copper shell after such a rock movement could lead to failure of the copper shell in a time perspective of tens of thousands of years. This requires further study.

Confidence

Process understanding: Our fundamental understanding of strength and deformations in the canister materials is good.

Models and data: Confidence in models and data is deemed to be sufficient for the calculations presented in the base scenario. The analyses may need to be expanded to include inhomogeneous material properties as well, which can occur as a result of e.g. casting defects in the iron insert. Calculations of creep movements also need to be revised as data on the actual canister material become available.

Conclusions

The long-term stability of the rock is judged in the base scenario to be such that no rock movements that could lead to canister failure will take place during the next 100,000 years. Rough calculations show that much larger shear stresses and stress levels than those that generally prevail in the Swedish bedrock at approximately 500 m depth would be required for creep movements or load changes caused by large-scale tectonic movements, individually or in aggregate, to lead to canister failure. There is nothing to indicate that a million-year perspective would not result in the same assessment.

This assessment applies to the three repository sites Aberg, Beberg and Ceberg, where conditions do not deviate significantly from conditions in the rest of the Swedish bedrock.

Nor do the groundwater or swelling pressures (evenly or unevenly distributed) which can occur in the base scenario give rise to loads that could threaten the canister's integrity in a hundred-thousand-year perspective.

The base scenario, and thereby its conclusions, does not cover earthquakes, see further section 3.5.

3.2.7 Chemical evolution

The chemical evolution is analyzed by means of an integrated treatment of all the chemical reaction and transport processes that have been identified in the system description. Here there are couplings to above all the hydraulic evolution in buffer and geosphere.

The interior of the canister is chemically isolated from its surroundings as long as the copper canister remains intact, and the emphasis is therefore on the evolution in geosphere and buffer/backfill, as well as on external corrosion of the copper canister. The Process Report shows that many processes are of no importance for the evolution in the base scenario, and they are therefore not dealt with.

The following descriptions are needed in the base scenario:

- The long-term evolution of the groundwater composition. An important requirement here is that the water at repository depth be oxygen-free.
- The chemical evolution of the buffer given the evolution of the groundwater.
- Canister corrosion.

Importance for safety: The chemical evolution of the repository may have a direct bearing on safety, since corrosion of the canister could influence its isolating capacity. Furthermore, buffer function is influenced by chemical changes. In the long term, the chemical properties of the groundwater, along with the properties of the buffer and the copper canister, determine how long buffer and canister will function.

Long-range evolution of groundwater composition

Based on site-specific measurements, a representative composition is determined for the groundwaters that exist today at repository depth for each site. The water at Aberg is saline, at Ceberg non-saline, while at Beberg both water types occur. The water on all sites is oxygen-free. Concentrations of numerous other substances of importance for canister corrosion, buffer stability, radionuclide transport etc. are given site-specifically.

Construction of the repository will disturb the water composition due to the fact that the flow conditions are altered when the host rock is partially drained and due to the introduction of oxygen and engineering and stray materials.

The original flow conditions are reinstated as the repository is resaturated after closure. Introduced organic material is expected to react with introduced oxygen, trivalent iron in the groundwater and possible sulphate in the groundwater. Introduced inorganic material, particularly concrete and steel, are judged to have a marginal effect on groundwater composition. After around 100 years, the groundwater is expected to have returned to its original composition on the three sites.

Based on measurement of the undisturbed groundwater composition, knowledge of hydrochemical processes and experience from model studies of long-term hydrochemical changes, both past and future, expert assessments are made of the groundwater's composition at closure and during different epochs in the future. The concentrations of important components are given as value ranges, site-specifically for each epoch.

The groundwater at repository depth is expected to be oxygen-free shortly after closure and for all future time. Oxygen in infiltrating surface water is as a rule consumed by biological processes as it passes through the soil layer. Furthermore, there is a very great potential for oxygen consumption in the minerals in the rock.

In about 2,000 years, ongoing land uplift is expected to transform the island of Aberg into a part of the mainland coast. The principal effects for the groundwater will be a slightly increased flow and a gradually diminishing salinity with increasing depth. In a ten-thousand-year perspective, the groundwater composition at Aberg is expected to approach that of today's non-saline water at Ceberg.

Beberg is currently in a transition from saline to non-saline water as a result of persisting effects of the historical climatic evolution, including land uplift. In a ten-thousand-year perspective, the composition of Beberg's groundwater is also expected to approach that of the non-saline Ceberg water.

At Ceberg, the long-term chemical evolution is not expected to alter the groundwater composition appreciably, with the premise in the base scenario that the present-day climate will persist.

Confidence

Process understanding: Our understanding of the fundamental reactions as well as the mechanisms of solute transport with the groundwater is good, see the Process Report.

Models: Models that deal with groundwater composition and evolution in different ways are not used directly for predictions in the safety assessment. The model set is constantly being developed and helps improve our understanding of the hydrogeochemical evolution on a repository site.

Data: Data uncertainty is generally great. In the first place, the present-day situation can only be observed at isolated points (boreholes) in the heterogeneous geosphere, and in the second place, the future evolution will be controlled by conditions outside the repository area and by residual effects of the historic climatic evolution, both of which are associated with great uncertainties. In recognition of these uncertainties, the future groundwater composition on a repository site must be specified as a range of values, in accordance with the account in the main report. Confidence that such ranges can be determined by systematically taking into account factors and uncertainties that influence the values is good.

Chemical evolution of buffer/backfill

Together with the influence of the surrounding groundwater, several processes contribute to the chemical evolution in the buffer/backfill, above all:

- ion exchange, where the montmorillonite's original content of sorbed sodium ions is exchanged for ions in the surrounding groundwater, primarily calcium,
- dissolution of impurities in the buffer, which, together with the influence of the site-specific groundwater, leads to changes in pH and redox conditions,
- chemical transformation of montmorillonite to non-swelling minerals, particularly through illitization.

The processes affect the ion content of the montmorillonite, the concentration of solid impurities, pore water composition and density in the buffer. This in turn affects the buffer's function.

Furthermore, possible erosion of the buffer due to removal of clay by the groundwater could reduce the density of the buffer in the long term.

Ion exchange and dissolution of impurities are modelled in an integrated fashion, where site-specific groundwater compositions and the two most important impurity minerals, pyrite and calcite, are included. The distribution of sodium/calcium montmorillonite, calcite concentrations, pHs and redox conditions in the buffer are calculated as a function of time for all sites.

Ion exchange proceeds fastest at Aberg, where the groundwater is calcium-rich and the flow rate is relatively high. Nearly complete ion exchange is expected to occur within a time horizon of 100,000 years, which in turn leads to a reduction of the swelling pressure from 7–8 MPa to 4–5 MPa. At Beberg the process is roughly 10 times slower, and at Ceberg the conversion is never complete since the sodium/calcium ratio in the groundwater is too high. A reduction of the swelling pressure to 4–5 MPa is of no importance for the function of the buffer.

The influence of the surrounding groundwater and the dissolution of calcite lead to changes in pH and redox conditions. The most striking changes are foreseen at Ceberg, where the pH is projected to rise to 10.7, later falling to just over 8. The changes do not affect the normal function of the buffer.

Pyrite dissolution requires that the groundwater contain oxygen and is therefore not expected to occur other than during the brief initial period when oxygen that has been introduced during construction and operation is consumed.

Sodium montmorillonite can be converted to non-swelling minerals, above all through **illitization**, which requires a supply of potassium. By assuming that only the potassium supply limits the scope of the process, an upper limit can be calculated for the conversion rate. At Aberg, where the supply rate of potassium is highest, 2.5 percent of the buffer is calculated to have been converted after one million years with this pessimistic estimate. Illitization is therefore expected to have a negligible effect on buffer function.

Present-day knowledge shows that **erosion** of the buffer is prevented if the groundwater contains sufficient concentrations of positive ions, e.g. calcium ions. The evolution of the groundwater composition described above shows that this is the case on all sites, both today and in the future. The scope of erosion in the long term with extremely ion-poor groundwaters may need to be further investigated.

In summary, the buffer is expected to retain a sufficiently high swelling pressure, a sufficiently high density and a sufficiently low hydraulic conductivity on all sites in a very long time perspective.

The backfill has not been analyzed in the same detail as the buffer. Furthermore, no adaptations of the backfill material to the site-specific groundwater compositions have been made in SR 97. The treatment of the long-term function of the backfill material needs to be refined for future safety assessments.

Confidence

Process understanding: Our fundamental understanding of all processes involved in the chemical evolution of the buffer is sufficient for satisfactory treatment in the safety assessment. The long-term effects of erosion under extreme conditions may need to be further investigated.

Models and data: Confidence in the models and data used for calculations of the chemical evolution of the buffer is deemed to be sufficient for the relatively rough model calculations required in the base scenario.

Canister corrosion

Copper is very stable in the environment in a deep repository. The only known copper corrodant that has been identified in deep Swedish groundwaters is sulphide. Initially, oxygen is also present in the buffer and the tunnel backfill, as is sulphate which can be converted to sulphide. Soon after deposition, small quantities of nitric acid could also conceivably be formed by radiolysis of the buffer's pore water.

Pessimistic rough calculations show that none of these factors threatens canister isolation, even in a million-year perspective. Nor has any mechanism that could lead to a local corrosion attack been identified.

Confidence

Process understanding: Our fundamental understanding of different mechanisms of copper corrosion in the deep repository environment is good, see the Process Report.

Models and data: In the safety assessment, quantitative estimations of copper corrosion are performed by means of rough calculations, which provide sufficient accuracy. Data uncertainties are above all associated with the availability of corrodants. The uncertainties are handled pessimistically in the safety assessment.

Conclusions

No long-range changes have been identified in the base scenario that contradict the conclusion that the groundwater at repository depth will remain oxygen-free in a million-year perspective.

The long-range composition of the groundwater in general, and its content of substances that could harm the canister or buffer in particular, can be estimated site-specifically by the use of ranges. The size of the range is determined by uncertainties in the present-day site-specific groundwater composition and in the long-term evolution of the flow situation and the hydrochemical conditions in the bedrock.

The range can be used to study the chemical evolution in buffer and canister, for example by pessimistically assuming that the most unfavourable value in the range will prevail. Such analyses show that the buffer is expected to retain a sufficiently high swelling pressure, a sufficiently high density and a sufficiently low hydraulic conductivity on all sites in a very long time perspective. The canister is projected to withstand the corrosion attacks to which it will be subjected in a million-year perspective with good margin.

The treatment of the long-range function of the backfill material needs to be refined for future safety assessments. The extent of erosion under extremely ion-poor groundwater conditions may also need to be studied further.

3.2.8 Summary: The base scenario in a time perspective

As a summary of the results of the analysis of the base scenario, the entire time sequence is summarized here, divided into three epochs.

The initial 100 years

The radiotoxicity of the fuel declines during this epoch to approximately 60 percent of its radiotoxicity at deposition.

Immediately after deposition, a heating of the entire repository begins, driven by the decay heat in the fuel. The maximum temperature on the outside of the canister, 90ºC, is reached after about 10 years. The temperature maximum at the boundary of the deposition holes is reached after about 20 years. On a larger scale, the temperature maximum at repository depth is reached after 90 years in Beberg (45ºC) and after 80 years in Ceberg (40ºC).

The buffer, which initially has a degree of saturation of about 80 percent, absorbs water from the surrounding rock as it heats. The time to full water saturation is some ten years or so and varies with the hydraulic conditions in the rock around the deposition hole. At the same time, the groundwater level above the repository is gradually restored.

In the final phase of the buffer's water saturation sequence, a swelling pressure is developed against the canister. The swelling pressure and the groundwater pressure together exert a total pressure of around 12 MPa against the canister, which is far below the mechanical load the canister can withstand. The swelling pressure may be unevenly distributed over the canister surface, especially during the water saturation phase. The mechanical stresses to which this gives rise in the canister are also far below the stresses the canister is designed to take. The thermal expansion of the host rock can cause millimetre-sized fracture movements around the deposition holes.

The hydrogeochemical evolution during the initial 100 years is characterized by a perturbation of the natural situation due to the fact that water in the region around the repository has been drawn in towards the repository as a consequence of the constant pumping-out of the groundwater during construction. Deeper-lying saline water can in this way be drawn up to the vicinity of the repository. The chemical conditions are also disturbed by the oxygen and engineering and stray materials brought into the repository during construction. Both organic and inorganic materials are expected to be consumed so that the groundwater composition approaches the original composition within 100 years.

One hundred to 10,000 years

The radiotoxicity of the fuel decreases during this epoch from 60 percent to approximately 0.6 percent of its radiotoxicity at deposition. The heating of the geosphere continues, and a heat wave is projected to reach the surface, although its impact will be negligible. In the case of a two-level repository in Aberg, the temperature maximum between the repository levels (55ºC) will be reached after about 450 years.

Hydraulically only small changes occur during this epoch. The buffer is saturated with water and the groundwater's flow in the geosphere is similar to the natural situation that prevailed before repository construction. With time, land uplift will affect the flow, especially on the island of Aberg, which is expected to become a part of the mainland in about 2,000 years. But the effects on the flow will be small.

The mechanical situation for buffer and canister is expected to be steady-state, since the buffer remains water-saturated. The heating of the geosphere leads to a build-up of stresses that are partially relaxed by thermal expansion. Certain fractures close and others open, but the effect is probably not great enough to cause fracturing. In major fracture zones, centimetre-sized thermally-induced movements may occur.

In the long term, changes in the flow conditions cause the saline water at Aberg and the mixture of non-saline and saline water at Beberg to change to a composition that increasingly resembles today's non-saline water at Ceberg.

In the buffer, the buffer's original content of sodium ions is exchanged for calcium ions in the groundwater. Calcite in the buffer is slowly dissolved.

The buffer has such a low hydraulic conductivity that transport of solutes, including canister corrodants, takes place entirely by diffusion. Corrosion processes in the copper shell have negligible consequences during this epoch.

The time after 10,000 years

In reality, major climate changes are likely during this epoch. The course of events under present-day climatic conditions is studied in the base scenario to be used as a basis for comparison to judge the effects of climate changes that are uncertain in both type and scope.

In 100,000 years, the radiotoxicity of the fuel declines to about 0.05 percent of its initial level and then remains on a par with that of the uranium ore mined to produce the fuel. There are still small amounts of radionuclides in the fuel that can move relatively easily through the repository's barriers if the canister should be damaged, and larger quantities of less mobile nuclides. The decay heat after 10,000 years has declined to less than one percent of its original value, and the temperature conditions in the repository system once again approach the natural situation.

Since today's climate persists according to the scenario definition, the hydraulic situation in the geosphere remains unchanged. The groundwater composition around the repository also remains unchanged.

During this epoch, the mechanical load on the repository rock is affected by slow, large-scale movements in the bedrock as well as by the rock's own long-range material properties, which can cause time-delayed deformations (creep movements). Rough calculations show that the effects of both these processes are negligible.

The effects of copper corrosion are very small, even in a million-year perspective.

The chemical evolution of the buffer is characterized by continued ion exchange and dissolution of calcite. In Aberg, a complete exchange from sodium to calcium is expected after several hundred thousand years. The effect will be a reduction of the swelling pressure to 4–5 MPa. At Beberg the process is about 10 times slower, and at Ceberg the transformation is never complete since the sodium/calcium ratio in the groundwater is too high. The reduction in swelling pressure is of no importance for the function of the buffer. Nor does the calcite dissolution or the influence of the surrounding groundwater cause any changes that affect the buffer's normal function.

3.2.9 Overall conclusions

The canister retains its isolating function during the evolution of the repository system in the base scenario.

The mechanical stresses on the canister from groundwater pressure, buffer swelling pressure and rock movements around the deposition holes are all far below what is needed to jeopardize canister isolation. The mechanical evolution in the host rock has been discussed in a hundred-thousand-year perspective, and there is no reason to doubt that a million year perspective would result in the same assessment.

Nor do the chemical stresses on the canister in the form of corrosion by oxygen or sulphide cause damage to the copper shell that would jeopardize isolation, not even on a timescale of a million years.

The assessment is based, among other things, on the requirements that the canister's surface temperature should be less than 100°C and that the water at repository depth should be oxygen-free. The former can always be achieved by a suitable deployment of the deposition holes or by adjusting the fuel content of the canisters. Oxygen has never been observed in deep Swedish groundwaters. Oxygen in rainwater is as a rule consumed effectively before it leaves the soil layer. Furthermore, there are microbes in the rock and minerals in both rock and buffer with a very great potential for oxygen consumption. In the base scenario, no long-term changes or processes have been identified which contradict the conclusion that the groundwater at repository depth will be oxygen-free in a million-year perspective.

The assessment of canister integrity is also based on the requirement that the buffer should function as intended, which among other things means that the buffer should have a sufficiently low hydraulic conductivity, a sufficiently high density and a sufficient swelling pressure. Processes such as ion exchange, mineral transformations or erosion do not cause in the base scenario any changes in the properties of the buffer that could jeopardize canister function in even a million-year perspective. The result is as expected, considering the fact that the buffer material is taken from a natural environment where conditions have for millions of years resembled conditions at repository depth in Swedish bedrock.

Our understanding of the long-term function of the backfill material and of buffer erosion under extreme hydrogeochemical conditions needs to be refined for future safety assessments.

3.3 Canister defect scenario

The premises for the canister defect scenario are the same as for the base scenario, except for one important point: A few canisters are postulated to have initial defects so that the isolation function can be said to be jeopardized already at repository closure. The reason for the defects could be small, undetected fabrication defects.

Otherwise, as for the base scenario, the situation in brief is that the repository is postulated to be designed according to specifications, and present-day conditions in the surroundings are assumed to persist. The evolution in and around the majority of canisters, which are assumed to be undamaged, is thereby expected to be the same as for the base scenario and is therefore not dealt with in the canister defect scenario.

A number of conditions that furnish input data to calculations of radionuclide transport are investigated in the canister defect scenario:

- The hydrochemical evolution in a defective canister, i.e. how water can enter a small defect in the canister shell and lead to corrosion with extensive gas generation and eventually to mechanical stresses which enlarge the initial defect.
- The chemical evolution in a defective canister, above all corrosion of the cast iron insert, fuel dissolution and dissolution of radionuclides.
- The hydraulic situation in the geosphere, which must be described much more accurately than in the base scenario.
- Transport processes for radionuclides in canister, buffer and geosphere.
- Radionuclide migration in the biosphere.

Furthermore, probabilities and sizes of defects in the canister's copper shell are discussed, along with criticality conditions in a defective canister.

Data from the above areas are used for calculations of radionuclide migration in canister, buffer, backfill, geosphere and biosphere. An attempt is made to determine both reasonable and pessimistic values for all input data, as well as statistical distributions in the few cases this is possible. The data set is used to formulate a number of calculation cases that result in estimates of doses to individuals in the vicinity of the repository. Probabilistic calculations are also carried out to obtain a risk measure that can be directly compared with the acceptance criterion for a deep repository.

All processes and dependencies that occur in the base scenario (Chapter 8) also occur in the canister defect scenario. In the case of canisters with penetrating defects, a number of additional processes also occur, most inside the damaged canister. In addition, a detailed description is given of processes that influence the transport of radionuclides in buffer, backfill and rock.

3.3.1 Initial canister defects

An estimation of the size and frequency of initial defects must be based on assumptions and reasoning. Statistically relevant data on defects and frequencies cannot be built up until experience has been gained from a large number of canisters that have been sealed and then inspected. Even such a body of data would be of limited value, since the canisters that are discovered to have defects would be discarded or repaired, and can therefore not be used directly to estimate the frequency of defective canisters which escape detection.

The fundamental reasoning in judging the size and extent of the initial defects is firstly that there are only a few events that could lead to an initial defect, and secondly that the defect must be small enough so that there is a reasonable likelihood that it will escape detection in a quality-control inspection. If a defect does occur, it is therefore most probable that it will be in the weld between the lid and body of the canister. The other welds in the canister shell are easier to inspect, since inspection can be performed both internally and externally and furthermore in a non-radioactive environment.

In the canister defect scenario, it is postulated as a reasonable case that one canister out of a total of about 4,000 has passed through quality inspection with a penetrating defect 1 mm^2 in size. In the pessimistic case, it is assumed that five canisters (i.e. about 0.1 percent) have such defects.

3.3.2 Radiation-related evolution, criticality

The radiation-related evolution is assumed to be essentially the same for a damaged and an undamaged canister, i.e. radiation levels in and around the canister are affected very marginally by a defect in the copper canister with accompanying water ingress.

One important question concerning the radiation-related evolution must be investigated in detail, however: Might the conditions in a damaged canister under any circumstances possibly be such that a fission process becomes self-sustaining? Here it is important to study different fuel types, burnups and hydraulic conditions inside the canister.

Premises

The criticality conditions have been calculated assuming the canister design described in section 2.3, for BWR fuel of type SVEA-64 with a mean enrichment of 3.6 percent U-235 and for PWR fuel of type F17x17 with a mean enrichment of 4.2 percent U-235.

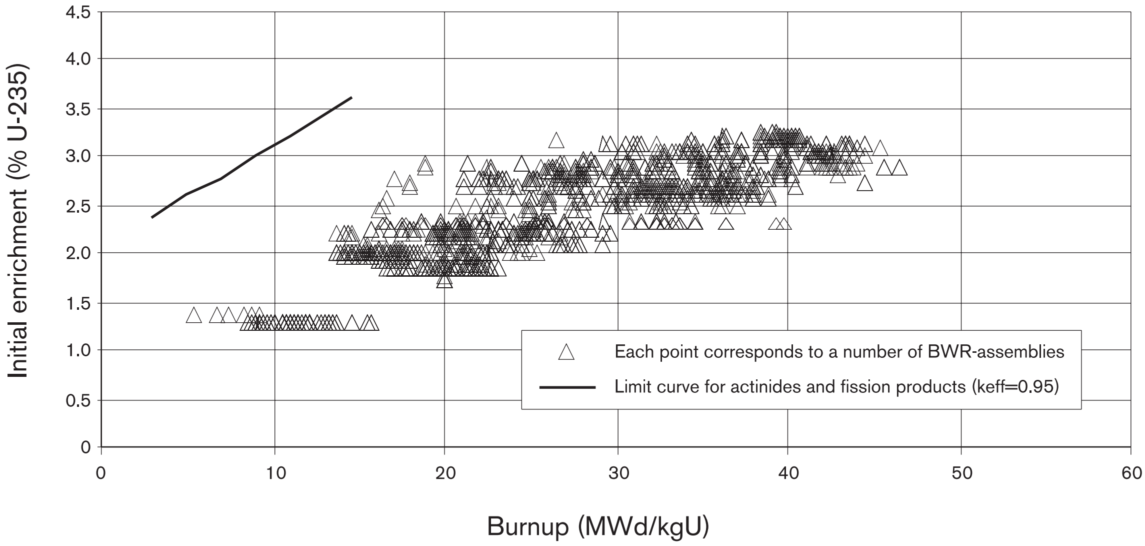

***Figure 3-6.** Limit curve and fuel data for BWR fuel.*

These fuels provide coverage from a criticality viewpoint of the fuel types expected to occur in an actual repository. The reference fuel in SR 97 (BWR, SVEA 96) is less prone to criticality.

The calculations are done for a situation where the fuel is postulated to be placed in canisters where the cavity in the insert has been filled with water and where the canisters are surrounded by bentonite externally.

Figure 3-6 shows a curve with combinations of enrichment and burnup that give acceptable criticality properties for BWR fuel. Fission products that absorb neutrons and thereby lower reactivity are included in the calculation.

The properties of all BWR fuel in CLAB as per 31 December 1998 are plotted in the figure. All fuel assemblies lie below the limit curve in the graph and can thereby be accepted from a criticality viewpoint for placement in the canister. The result for PWR fuel is similar.

Before the calculations were done, uncertainties in burnup determination were analyzed. Effects of uneven burnup in the fuel assembly, of removed fuel rods, varying spacing between channels, gap width between assembly and channel, eccentric placement, and cavities and porosities in the insert were also analyzed. The calculations were done with margin for uncertainties in all these factors.

Long-term changes in the fuel (changes in the inventory as a consequence of radioactive decay) and the canister (changed environment as a consequence of corrosion) counteract the occurrence of criticality.

Conclusions

The analyses show that the spent BWR and PWR fuel that is present in CLAB and that which comes from the Swedish nuclear power plants can be disposed of in the canisters with good margin to criticality, even if the canisters should for some reason be filled completely or partially with water. Future changes in isotope composition, material or geometry are not projected to reduce the margin to criticality.

3.3.3 Hydromechanical evolution in defective canister

The hydraulic evolution in a canister with a damaged copper shell underlies all essential processes that distinguish the evolution of a damaged canister from that of an undamaged one: Ingress of water is a prerequisite for corrosion of the cast iron insert, which in turn gives rise to generation of hydrogen gas. Water is also a prerequisite for corrosion of the fuel's metal parts, fuel dissolution and radionuclide transport.

Due to the fact that the corrosion processes both consume water and generate hydrogen, strong couplings exist between the chemical and the hydraulic evolution, which must therefore be described in a single context. Certain aspects of the mechanical evolution must also be dealt with in parallel with the hydrochemical processes.

The processes are important for the function of a repository. In order for radionuclides to be transported out of the spent fuel, there must be a continuous water pathway between the fuel and the groundwater in the rock, with the exception of nuclides that are transported with gas. Due to consumption of water in the canister, it may take a very long time before such a water pathway forms.

The description is based on two new model studies of the hydromechanical evolution in a canister with a damaged copper shell.

Hydraulic evolution in canister

In the event of a penetrating defect in the canister's copper shell, water can be driven through the buffer and into the canister by the difference between the internal gas pressure in the canister and the groundwater pressure. When water comes into contact with the iron insert, it will corrode. Iron corrosion consumes the intruding water, at the same time as hydrogen gas is generated and the gas pressure in the canister increases. The pressure differential across the buffer is thereby reduced and the rate of water inflow decreases.

Model studies indicate that water entering a canister will be consumed and dry conditions will prevail during long periods. The course of the process is, however, dependent on the corrosion rate and the shape of the breach in the shell.

Water ingress via diffusion

When the gas pressure in the insert reaches the groundwater pressure, the inflow of water to the canister will cease. Water can then continue to enter by diffusion, thus sustaining corrosion. Diffusion is very slow; with a 1 mm^2 circular hole and a corrosion rate of 0.1 μm/y, the inflow is about 10^{-5} l/y.

Mechanical effects of corrosion products

When the insert corrodes, a layer of magnetite will be built up between insert and shell. Magnetite has a lower density than iron, so the corrosion products will exert a pressure between insert and shell. If the corrosion occurs locally around the defect in the copper shell, the calculations show that the shell is severely deformed around the defect without the defect itself becoming enlarged.

The shell is calculated to fail after a strain of approximately 20 mm. The density ratio between magnetite and iron is 2.1:1, and with a corrosion rate of 0.1 μm/y the time to failure can be calculated to be approximately 200,000 years.

If corrosion takes place over the entire outer surface of the insert, the pressure from the corrosion products causes strain of the whole copper shell. In a long time perspective with global corrosion, the build-up of corrosion products causes the greatest strains around the canister lid. After approximately 200,000 years of global corrosion, the lid is expected to come loose.

To determine how the global corrosion affects buffer and rock, the mechanical stresses in canister, buffer and 10 cm of the rock have been calculated. When the insert has corroded completely, the radial strain in the copper shell is calculated to be 55 mm. The properties of the buffer enable it to absorb most of the stresses from the canister without transmitting them to the rock in this situation.

The maximum compressive stresses in the rock are calculated to be around 4 MPa, and the tensile stresses around 1.4 MPa. The compressive stresses are judged not to affect the rock. The tensile stresses could cause local cracks in the lower corner of the deposition hole.

Gas transport through buffer

In the water-saturated state, the buffer is impenetrable to flowing gas and a gas pressure is therefore expected to build up in the canister cavity. The gas can dissolve in water and diffuse through the buffer out to the rock, but the transport capacity of the buffer is not sufficient to remove gas generated by either global or local corrosion.

Several experiments have shown that bentonite does not allow gas to pass until the pressure in the canister exceeds the sum of the swelling pressure and the groundwater pressure, i.e. about 12–14 MPa. When the pressure reaches this value, a transport pathway is formed through the buffer and gas is released. The pressure falls and gas production determines the further course of events:

- If the pressure falls to a sufficiently low value, the transport pathway closes. This so-called "shut-in pressure" is dependent on the swelling pressure. At normal swelling pressure, 7–8 MPa, the shut-in pressure is 3–5 MPa, according to very preliminary estimates. After that, gas once again migrates solely by diffusion, see Figure 3-7. If gas production continues long enough, a cycle is obtained with successive gas pulse releases and pressure build-ups.
- If, on the other hand, gas production is high enough to sustain a higher pressure, the gas transport pathway is expected to remain open.

The buffer's gas transport capacity is the subject of investigations.

Sequence of events

Based on the model studies of different hydromechanical regimes, a reasonable sequence of events with a 1 mm^2 canister defect and a corrosion rate of 0,1 µm/y can be sketched:

Time up to about 11,000 years: Water is expected to flow into the canister driven by the pressure differential between the groundwater in the rock and the gas in the canister cavity. The corrosion rate is sufficiently high to consume all water. After about 11,000 years, the pressure in the canister is projected to reach 5 MPa, after which water is transported into the canister by diffusion of water vapour.

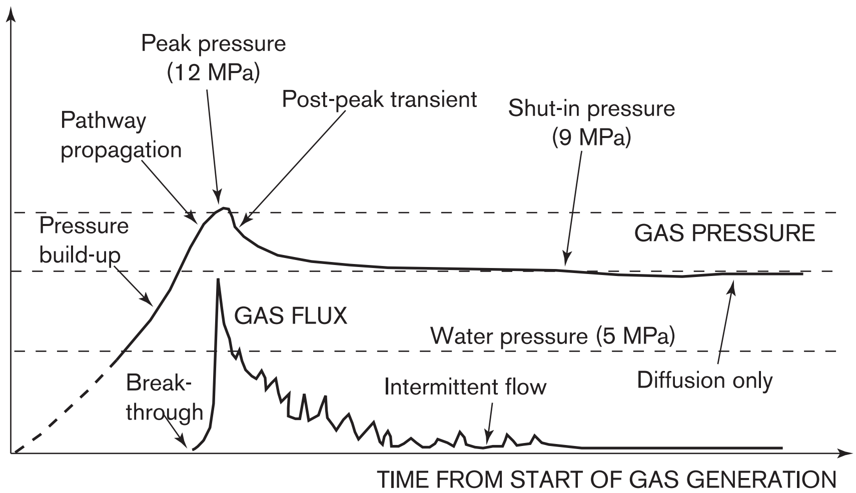

Figure 3-7. *Temporal course of gas transport through bentonite. The timescale in the figure is relative and dependent on the rate of pressure build-up.*

11,000 to 18,000 years: Water diffuses in around the defect and corrosion is only expected to occur locally in a radius of about 5 cm around the defect. After about 18,000 years, the corrosion products are projected to have filled the 2 mm wide gap between insert and shell, and the shell will begin to expand. Gas production is low in this time interval, since the inward transport of water is slow.

18,000 to 200,000 years: The corrosion products around the defect are expected to expand the copper shell. After approximately 200,000 years, the copper shell is projected to fail and a hole as big as the corrosion area will be created. Gas production is still low in this time interval.

200,000 to 400,000 years: When a larger hole has formed, water transport into the canister will increase once again, and it is probable that the entire surface of the insert will corrode. At this point it is not impossible that gas production will reach the maximum value of $1.5 \cdot 10^{-2}$ m^3/y (STP). The maximum pressure increase will then be about 1 MPa in 700 years, which means that the gas can probably not escape by diffusion and gas release can be expected in the buffer. The strains will now be greatest around the canister lid, and after about 400,000 years the lid is expected to come loose.

400,000 to 700,000 years: The global corrosion continues, and after appoximately 700,000 years the insert has corroded through to the fuel channels. The strain in the body of the copper shell is still less than 20 percent, which means it is not expected to crack. After approximately 700,000 years, gas production is projected to decline due to the fact that nearly all iron has been consumed and the gas releases through the buffer cease. The buffer has been deformed locally at the bottom of the deposition hole. The rock is mainly subjected to compressive stresses and is not judged to be damaged. The tensile stresses that occur around the deposition hole are very local.

Data for calculations of radionuclide transport

The hydromechanical evolution in a damaged canister furnishes data for a) the time it takes for a continuous water pathway between the fuel and the outside of the canister to be formed, and b) the time it takes for the internal evolution to have caused the small initial defect to suddenly grow to a larger defect. Radionuclide transport begins when the continuous water pathway has formed, but is restricted by the small initial defect. When the defect has grown larger, it is assumed that the canister no longer offers any resistance to radionuclide transport. Both reasonable and pessimistic estimates are needed for both times.

In accordance with the above account, 200,000 years is chosen as a reasonable value for the time when the initial defect grows to a larger defect. Up to this time, no liquid water is expected to be present in the canister and no radionuclide transport occurs in the reasonable case. When the initial defect grows into a larger defect, it is assumed that water can also enter freely, and 200,000 years is therefore the reasonable time when a continuous water pathway is expected to form.

It is pessimistically assumed that a continuous water pathway is formed after only 300 years. The small initial defect is then assumed to run around the entire circumference of the welded joint, whereby the influx of water is much faster than in the reasonable case with a small circular hole. The continuous water pathway permits radionuclide transport out through the small initial defect. It is pessimistically assumed that the corrosion rate of the iron is 1 µm/y, which gives full defect growth in 20,000 years.

Uncertainties

The details in the evolution sketched above must be regarded as very uncertain. The only data utilized in radionuclide transport calculations are, however, the time it takes to form a continuous water pathway and the time when the canister is assumed to lose its transport resistance.

The former is based in the pessimistic case solely on an analytical expression for inflow of water through a hole where the shape of the hole and the hydraulic conductivity of the buffer are pessimistically chosen, and where inflowing water is pessimistically assumed not to be vaporized.

The time it takes for the little defect to grow to a larger defect and for the canister to lose its transport resistance can be pessimistically bounded upward by assuming that the time is solely dependent on the corrosion rate with an unlimited water supply and the canister's strength. By choosing a pessimistic value for the corrosion rate as well (1 µm/y), a time is obtained which ought to be very pessimistic.

3.3.4 Chemical evolution in defective canister

The chemical evolution in a damaged canister differs radically from that in an intact canister in that intruding water gives rise to several important chemical reactions, chiefly:

- corrosion of the cast iron insert,
- corrosion of the fuel's Zircaloy cladding and other metal parts, releasing radionuclides in these parts,
- dissolution of the fuel matrix with release of radionuclides, and
- release of immediately accessible radionuclide fraction.

The composition of the intruding water and the ensuing reactions together determine the chemical environment in the canister. The chemical environment in turn determines how released radionuclides speciate, i.e. which chemical form they assume, and thereby also to what extent they occur in dissolved form, accessible for transport, or are precipitated in the canister. Speciation is of decisive importance for radionuclide migration from a damaged canister.

Corrosion of the cast iron insert

Groundwater at repository depth is oxygen-free. The cast iron insert will therefore corrode anaerobically, with hydrogen generation and magnetite formation. A magnetite layer is expected to be built up on the iron surface. When the layer has reached a thickness of 0.7–1 µm, the corrosion rate is expected to be around 0.1 µm/y, regardless of the continued growth of the magnetite layer and of whether the water occurs as a liquid or a gas saturated with water vapour. The value is used above all in the above analysis of the hydromechanical evolution.

Corrosion of metal parts and cladding tubes

The fuel's metal parts, which contain small quantities of radionuclides, are highly corrosion-resistant, but present-day knowledge and available data are not sufficient to quantify corrosion in the safety assessment. It is therefore pessimistically assumed that the entire inventory of radionuclides in the metal parts is immediately released when a continuous water pathway forms and radionuclide transport becomes possible.

The same applies to the Zircaloy cladding tubes that surround the fuel pellets. Here the transport resistance of the tubes is also pessimistically neglected.

Dissolution of fuel matrix

The majority of radionuclides in the fuel lie embedded in the fuel matrix of uranium dioxide and cannot be released until the matrix has been dissolved or transformed. A description of the dissolution/transformation of the fuel matrix is therefore needed in a safety assessment.

Under normal conditions in the repository (reducing environment, neutral to alkaline pH), uranium dioxide has very low solubility in water. If solubility is assumed to be the limiting factor, dissolution of the fuel matrix will proceed very slowly. Based on this, a solubility-limited model for the release of radionuclides from the fuel can be formulated.

In addition to this dissolution mechanism, it is also conceivable that oxidants formed by radiolysis of water around the fuel matrix could cause transformation of the fuel so that embedded radionuclides are released. Based on this, a model for fuel transformation resulting from radiolytic oxidation can be devised. This makes a further contribution to fuel dissolution, since the former mechanism always occurs.

The large quantity of reducing species in the canister, above all Fe(II) and Fe(0), could conceivably "neutralize" the effects of the oxidants formed by radiolysis. However, such a mechanism has not yet been experimentally or theoretically proved. In SR 97, a model is therefore used that assumes that the fuel matrix is dissolved as a consequence both of its "own" solubility and of the oxidants produced by radiolysis of water.

An earlier model for radiolytic oxidation has been further refined. The new model quantifies:

- radiolysis processes in the water between fuel and cladding,
- a series of reactions between different radiolysis products in the water and between radiolysis products and dissolved hydrogen from corrosion of the insert, and
- reactions between oxidants and the uranium dioxide, i.e. the direct cause of fuel dissolution.

The model calculations are performed with a standard program for modelling of radiolysis. Simulations with the model show that the system quickly reaches a steady state where the dissolution rate is determined by the partial pressure of the hydrogen in the canister and the dose rate.

With reference values of all relevant parameters, it is calculated that a fraction of 10^{-8} of the fuel is dissolved every year. The hydrogen concentration is of crucial importance for the result: Dissolved hydrogen reacts with OH^- to form atomic hydrogen, which consumes oxidants that could otherwise attack the fuel. A pressure of 0.5 MPa is sufficient for the limiting effects of the hydrogen to be substantial in the calculations.

The result is also governed by numerous other factors. The manner in which dose rate and the rate of the oxidants' reaction with the fuel affects the result is shown by means of variation analyses.

A dissolution rate of 10^{-8}/y is used in radionuclide transport calculations as a reasonable value. No pessimistic value of the dissolution rate has been chosen. The uncertainties surrounding fuel dissolution in terms of understanding, models and data are great. However, the dissolution rate is of limited importance in the safety assessment, for several reasons:

- Some nuclides in the fuel accumulate on the surface of the fuel and are therefore released even if the matrix is stable. I-129 and Cl-36 are examples of long-lived such nuclides.
- Other nuclides are present in the structural parts of the fuel and are released as they corrode, which is judged to be a faster process than matrix dissolution. Ni-59 and Ni-63 are present in the structural parts.
- Several matrix-bound nuclides, such as plutonium, have very low individual solubilities and precipitate in solid phases if the matrix is dissolved rapidly. This results in a very slow release of such nuclides.

The importance of fuel dissolution is illustrated in the radionuclide calculations by a special calculation case where dissolution is assumed to be immediate.

The model used for the SKB 91 safety assessment greatly overestimates the oxidation rate for the fuel. Based on the assumptions in SKB 91, the long-term fuel dissolution rate is such that complete dissolution can be expected in about 170 million years. In the long term, conditions in the fuel should be comparable to those at the uranium deposit at Cigar Lake, which is still largely intact after more than a billion years.

The model used in SR 97 is an attempt to give a more complete and realistic description of the radiolytic oxidation of the fuel. If the assumptions in the model were to be applied to Cigar Lake, this would result in complete oxidation in 100 million years. This suggests that models with radiolytic oxidation overestimate oxidation over long periods of time.

Chemical speciation of radionuclides, solubilities

In preparation for SR 97, a major effort was undertaken to determine solubilities and the uncertainties associated with them:

- Evaluation of the importance of the natural composition of the groundwater and the relative importance of different species, in order to get an idea of which complexes and solid phases will dominate.
- A literature survey of concentrations of (stable isotopes of) the important radioelements, in order to determine the reasonableness of the calculated solubilities.
- Survey of available data from fuel leach tests of concentrations of non-natural radioelements to get an idea from there as well of the reasonableness of the calculated solubilities.
- Calculation of solubilities for all three reference waters in SR 97, plus an assumed bentonite pore water.
- Comparison of the calculated results with natural concentrations, concentrations in leachates and results from other safety assessments.
- A sensitivity/uncertainty study to determine which factors are important for the solubility of each radionuclide.

Based on the study, site-specific reasonable and pessimistic data have been chosen for the nuclide transport calculations. The pessimistic values have been selected on the basis of the results of the variation analyses.

“Reliable” pessimistic solubility-limiting phases have been chosen throughout in the solubility calculations in SR 97. The actual concentrations of radionuclides in a water-filled canister can in many cases be several orders of magnitude lower than those used in the radionuclide transport calculations. Sensitivity analyses with respect to the composition of the groundwater show that the solubilities are in most cases relatively insensitive to even substantial changes in water composition. This is important, since the groundwater composition in the repository will change during its lifetime.

Solubilities are calculated for different species in the reference waters for Aberg, Beberg, Ceberg and for a special bentonite water. Sensitivity to variations in pH, Eh, carbonate content and temperature are also calculated.

Reasonable values for the solubilities are selected from the solubilities that have been calculated for a given reference water, but if the solubility of an element is higher in the bentonite water, this value is chosen as the reasonable one.

In view of the uncertainty regarding the future chemical environment inside the canister, the pessimistic solubilities are chosen by choosing for each element the highest solubility that has been calculated for any of the different water compositions. If the sensitivity calculations show that the solubilities could be even greater, these values are chosen.

3.3.5 Hydraulic evolution in the geosphere

The groundwater movements and their changes in time are determined by the hydraulic properties of the geosphere and the conditions in the surroundings, above all precipitation, which is in turn dependent on the climate. Since these factors are postulated to be the same as in the base scenario, the same hydraulic evolution as in the base scenario is also expected, i.e.:

- an initial transient phase where the geosphere is resaturated with water after repository closure,
- followed by a long-term phase where conditions resemble the undisturbed, natural conditions that prevailed before the repository was built. These conditions are expected to be essentially unchanged over a long period of time, with the exception of known trends such as land uplift.

Thus, the description of the hydraulic evolution on the regional scale for the three sites in the base scenario is valid for the canister defect scenario as well.

A more detailed description of the long-term phase is required on a local scale (up to about one kilometre) for the canister defect scenario than was given in the base scenario. The higher resolution is important to provide a good description of possible radionuclide transport from the repository.

The model HYDRASTAR is the main alternative for calculations on the local scale. HYDRASTAR is a finite difference model for stochastic continuum simulation of groundwater flow. In the model, the rock volume is divided into elements that are assigned hydraulic properties based on a three-dimensional hydrogeological description of rock mass and fracture zones. The description is in turn based on observations and measurements. The hydrogeological description has statistical aspects, and the elements are also assigned hydraulic properties stochastically.

A large number of realizations are used to shed light on uncertainties in a given calculation case. In each realization and for each element, the hydraulic properties are selected randomly from given distributions. The stages that lead from actual measurements to input data to HYDRASTAR include several rounds of expert assessments and interpretations.

The elements in HYDRASTAR are cubes with a side of 25 metres for Aberg and 35 metres for Beberg and Ceberg. The mean values of hydraulic conductivity for the rock mass in Aberg and Beberg are around two orders of magnitude higher than for Ceberg. The same applies to the fracture zones.

In the model, the tunnel positions are deployed in the hypothetical repository and the backfilled tunnels are assumed to have a hydraulic conductivity of 10^{-10} m/s, while the excavation-disturbed zone (EDZ) is neglected. With the element sizes used, the effect of neglecting the EDZ is minimal.

More than a hundred representative canister positions are deployed in the model, and the following is calculated for each one of these in each realization:

- flux (Darcy velocity) at repository depth $[m^3/(m^2 \cdot y)]$,
- advective travel times from canister positions to the boundary between geosphere and biosphere (years),
- coordinates for exit points at the ground surface.

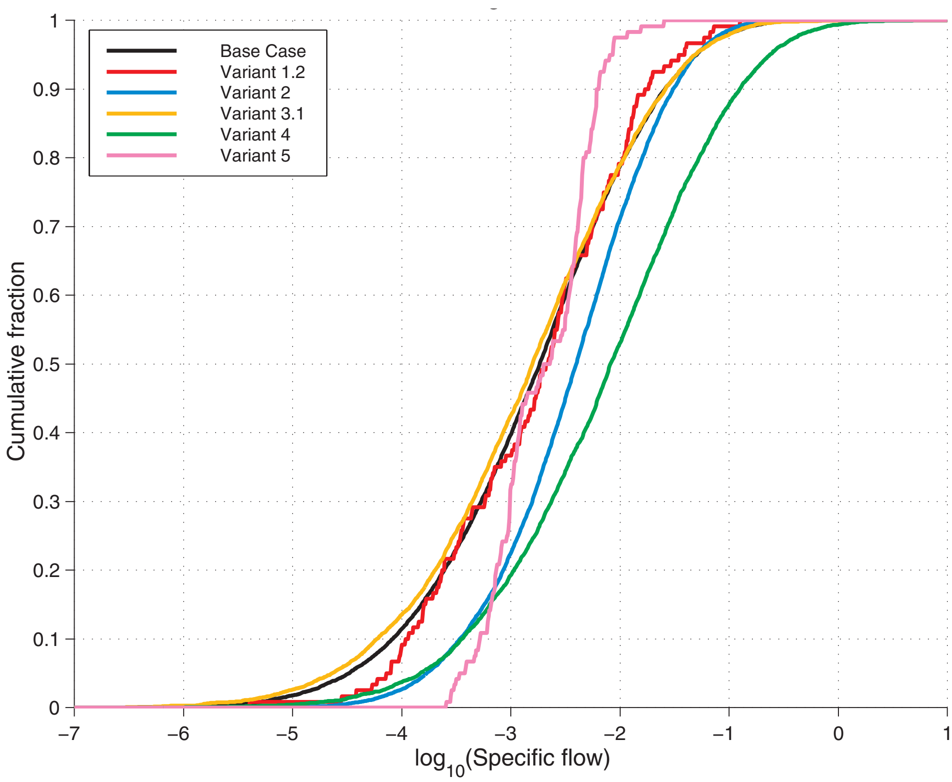

Figure 3-8. *Groundwater flux for the base case and variants at Aberg.*

The advective travel time is a theoretical quantity that is used to transfer calculations results from hydromodels to transport models. There the travel times can be used to calculate the time for transport of solutes from repository depth to the surface. These times are often several orders of magnitude longer than the advective travel times.

The boundary conditions for the calculations are taken from the results of the regional modelling that was described in the base scenario. The calculation result is then used as input data for the calculation of radionuclide transport in the geosphere, see section 3.3.8.

Results Aberg

For Aberg there is a base case where data have in principle been chosen to provide as reasonable a description as possible. In addition, five variants have been defined by a group of experts to shed light on questions concerning models and properties of the system:

1. Alternative boundary conditions.
2. Upscaling of hydraulic conductivity.
3. Anisotropy in statistical description of hydraulic conductivity.
4. Hydraulic conductivity inferred from measurement data.
5. Deterministic simulation with constant conductivity within each individual domain for rock mass and fracture zones.

Figure 3-8 shows the results for groundwater flux for the base case and variants. The results represent statistics for all realizations and all canister positions.

Figure 3-8 shows that the base case and variants are very similar as regards both the location and spread of the distributions. The width of the individual distributions in Figure 3-8 represents the uncertainty given by the natural variability combined with the random canister position. The figure shows that this combination of uncertainties is much greater than the range of variation between the variants. An overall conclusion is therefore that the effects of the spatial variability dominate over the uncertainties that stem from assumptions regarding the properties of the system.

Analysis of individual canister positions gives narrower distributions. Some of the range of variation in Figure 3-8 thus stems from differences between different starting positions. The analysis of individual canister positions also shows that different canister positions in the repository have systematically different mean travel times and fluxes. Such information could be used to optimize the repository to some extent by avoiding unfavourable positions. This has not been done in SR 97.

Results Beberg

For Beberg, a base case and four variants have been analyzed:

1. Boundary conditions from a regional model with salinity effects.
2. Alternative conductive structures (more fracture zones).
3. Alternative hydrogeological interpretation.
4. Deterministic simulation with constant conductivity within each individual domain for rock mass and fracture zones.

The overall result for Beberg as well is that the results from the base case and variants are very similar. The results of the base case are also comparable to the results from the SKB 91 safety assessment where Beberg was analyzed.

Results Ceberg

For Ceberg as well, a base case and four variants have been analyzed:

1. Contrast in hydraulic conductivity between domains for rock mass and fracture zones increased by a factor of 100.
2. Alternative conductive structures.
3. Increased variance in hydraulic conductivity.
4. Deterministic simulation with constant conductivity within each individual domain for rock mass and fracture zones.

The results show that the base case does not cover all variants as well as at Aberg and Beberg. The differences between the variants are, however, still less than the difference within the base case and within individual variants.

Comparison between the sites

Groundwater fluxes are compared in the base cases for the three sites in Figure 3-9. Aberg and Beberg have mean values of the same order of magnitude, while Ceberg is approximately two orders of magnitude lower. Aberg has the greatest spread, while Ceberg has the smallest.

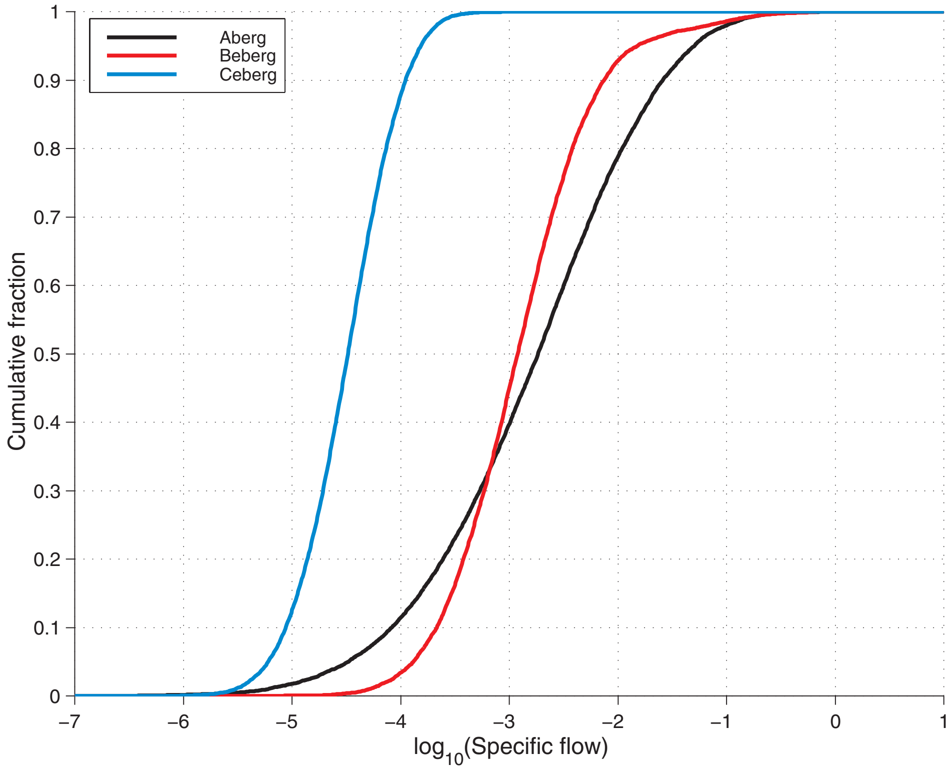

***Figure 3-9.** Groundwater flux for the base cases in Aberg, Beberg and Ceberg.*

Uncertainties

The purpose of the local models is to provide an understanding of the hydraulic situation at the site. The models are burdened with uncertainties caused by:

- the chosen model concept,
- spatial variability, and
- incomplete data on the site's geological structures and incomplete knowledge of other governing properties.

Conceptual model uncertainty: To analyze conceptual model uncertainty, not only HYDRASTAR but also the discrete fracture network model FracMan/MAFIC/PAWorks and the channel network model CHAN3D have been used for Aberg.

All models give equivalent travel times and fluxes on repository level. The results for median travel time and median flux in particular are very similar. Even though the spread in HYDRASTAR is greater than in FracMan and CHAN3D, the models give similar values for the shortest travel times and the highest fluxes. The locations of the exit points are very similar in the different models.

The results indicate that the problem premises rather than the chosen model determine the result, i.e. that the conceptual uncertainty is low. HYDRASTAR's base case and variants in Figure 3-8 cover virtually the entire range of results from the different models.

Spatial variability: It is shown for all sites that the heterogeneity of the geosphere, in combination with the randomly selected position of a damaged canister, gives the greatest uncertainties in travel times and fluxes.

Uncertainties in data and properties: Uncertainties stemming from incomplete data and knowledge of the properties of the system are analyzed by comparing a base case with a number of variants. The results show that the differences between different variants are usually small. Salinity effects at Beberg constitute an exception where the dominant flow direction is greatly affected by salinity effects.

Separate studies show that the quantities that are transferred to radionuclide calculations are only changed marginally by land uplift as well.

The calculated differences in travel times and fluxes between the sites can be explained by differences in hydrogeological conditions and data that have been used in the calculations. Measured differences between the sites are probably due to a combination of actual physical differences and the fact that different investigation methodologies have been used for different sites.

Coming work

The option of deselecting deposition positions based on the results of hydrocalculations has not been exploited in SR 97. Such a procedure would require both a refinement of methods and criteria and hydromodelling on a more detailed scale. This will be considered in future revisions of SKB's RD&D-programme.

3.3.6 Transport processes in the repository

The processes that can lead to dissolution of radionuclides in the water in a damaged canister have been discussed in previous sections. Radionuclides dissolved in water can be transported in the interior of the canister, principally by diffusion, and reach the buffer via the defect.

After water saturation, radionuclide transport in the buffer is expected to take place solely by diffusion in the buffer's pores, possibly also on the surfaces of the clay particles. Neither advection nor colloid transport occur due to the properties of the buffer. Radionuclides can be sorbed to the surfaces of the montmorillonite. A crucial factor for this is the chemical form of the radionuclide, which is determined by the chemical environment in the buffer by the process of speciation.

In the rock, radionuclides can be transported by the flowing groundwater, advection. Diffusion can also be important under stagnant conditions. An important aspect of this is matrix diffusion, i.e. radionuclides diffuse in the stagnant water in the micropores in the rock and are thereby retained and transported more slowly than the flowing water. The timescale for advection in relation to the timescale for matrix diffusion determines the relative importance of the latter process. Sorption, where radionuclides sorb (adhere) to the surfaces of the fracture system and the rock matrix, is also crucial for radionuclide transport. Matrix diffusion and sorption are the two most important retention processes for radionuclides in the geosphere. Another factor that can be of importance for retention is sorption on colloidal particles and transport with them. The chemical environment in the water determines what speciation (chemical form) the radionuclides will have, which is crucial particularly for the sorption phenomena. Some nuclides can be transported in the gas phase.

Finally, radioactive decay influences the groundwater's content of radionuclides and must therefore be included in the description of transport phenomena.

Our knowledge of all these processes has been evaluated in the Process Report, and reasonable and pessimistic values for the data required to quantify the processes are proposed in the Data Report.

3.3.7 Biosphere

Migration of radionuclides from the repository to man and nature in the vicinity of the repository is analyzed in the canister defect scenario. For this reason, a much more detailed analysis of biosphere conditions is required than for the base scenario.

Compared with the repository system (fuel, canister, buffer/backfill and geosphere), the biosphere is considerably more heterogeneous, complex and changeable. It is therefore difficult to carry out as strict and exhaustive a process description for the biosphere as for other parts of the system. Radionuclide turnover in the biosphere is, however, controlled by a limited set of processes that can be described in general terms.

The quantitative description is dependent on which ecosystem the processes take place in. To quantify radionuclide turnover for the safety assessment, the migration in a number of typical ecosystems is first calculated for uniform radionuclide releases to them, then the dose to humans is estimated. The calculations give an ecosystem-specific dose conversion factor (EDF) for each ecosystem and nuclide. The EDF expresses the continuous dose load (Sv/y) to humans in the surroundings given by a continuous radionuclide release to the ecosystem (Bq/y) with a duration of 10,000 years. The unit for the EDF is thereby Sv/Bq.

For application in the safety assessment's calculations, the three repository sites are divided into smaller areas and each area is classified as one of the typical ecosystems. Site-specific and time-dependent releases of radionuclides to the biosphere can then be converted to estimates of doses to man.

Calculation of ecosystem-specific dose conversion factors (EDFs)

In SR 97, radionuclide migration and dose load are modelled in a number of typical ecosystems:

- coast and archipelago area,
- lake,
- running water, stream, river,
- wetlands, peat bogs,
- agricultural land,
- well.

Forest has been assumed to be equivalent to peatland, which is judged to be pessimistic.

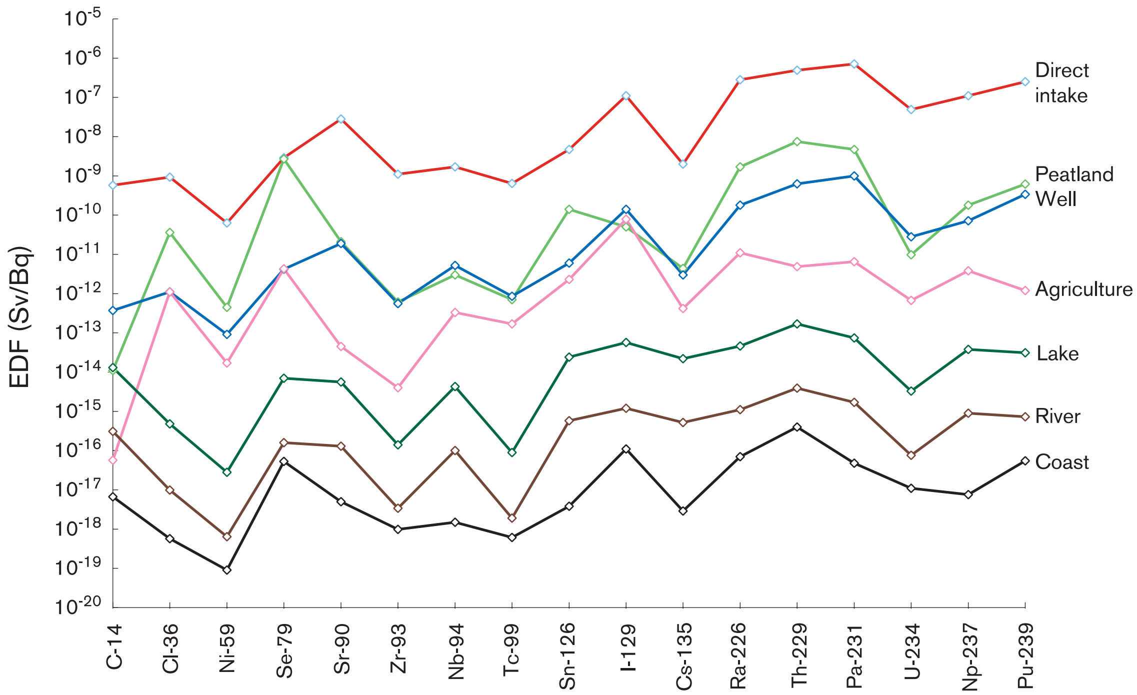

***Figure 3-10.** EDFs for a selection of radionuclides in different typical ecosystems. Direct ingestion refers to the dose conversion factor for ingestion via food without dilution.*

EDFs

The results of calculations of the ecosystem-specific dose conversion factors for well, agricultural land, peatland, coast, lake and river are shown in Figure 3-10. The figure also shows dose conversion factors for direct ingestion of the relevant nuclide with food.

The figure shows that the EDFs for many typical ecosystems differ by a factor that is largely independent of the radionuclide. Moreover, the shape of these curves resembles that for direct ingestion via food without dilution. The reason for the similarities is that varying dilution is the factor that distinguishes the modules from each other the most, and this factor affects all nuclides equally. In peatland and agricultural land, certain radionuclides accumulate over a prolonged period and dilution is slower, which explains why these ecosystems depart from the pattern.

Description and classification of typical ecosystems

Aberg: The typical ecosystem for archipelago and open coast dominates today. Forest-land and peatlands also comprise a large portion of the ecosystems. Small parts of the area are agricultural land, while lakes are lacking. There are three wells in the area today with a mean capacity of 300 litres per hour.

Land uplift is expected to change the situation, and in 2,000–5,000 years peatland will probably be the dominant ecosystem.

Beberg: Forest is the dominant ecosystem, followed by wetlands and agricultural areas. There are also lakes and watercourses. In addition, there are four wells with a mean capacity of 1,000 litres per hour. Since forestland is pessimistically replaced by peatland in the modelling, the EDF for peatland will dominate.

Land uplift and drying-up of lakes are not expected to alter the present-day situation appreciably.

Ceberg: The site is drained by two major watercourses. Small areas of cultivated land occur along one of them. Peatlands and forests are the dominant ecosystems. As a result, the peat ecosystem dominates in the area. Watercourses and agricultural land also occur frequently. There are two wells with an average capacity of 500 litres per hour within the area.

Land uplift is not expected to affect the distribution of the dominant ecosystems at Ceberg.

Which acceptance criterion?

In calculations of EDFs, it is assumed for the typical ecosystems peatland, agriculture and well that releases of radionuclides to the biosphere are dispersed within a local area of roughly the same size as the subareas as were used to subdivide the biosphere on the three sites (250 m x 250 m). It is assumed that residents produce all their food and get all their drinking water within this area.

For at least three typical ecosystems, the calculations thus include the most exposed individuals within a large region. The acceptance criterion which the calculation results should be compared with when these EDFs are used is thus a risk of 10^{-5}, i.e. a risk for the most exposed individuals in a population. Expressed in dose, this risk is equivalent to $1.5 \cdot 10^{-4}$ Sv/y for an exposure that is certain to occur.

For the typical ecosystems coast, lake and running water, it may in some cases be large populations that are exposed, which makes it more reasonable to compare with a risk of 10^{-6}, which is equivalent to a dose of $1.5 \cdot 10^{-5}$ Sv/y for an exposure that is certain to occur.

Uncertainties

The employed methodology of dividing the sites into subareas enables the uncertainties in the near-surface ecosystems to be separated into the question of which ecosystem will be the discharge area, and uncertainties in input data for the calculation models for each typical ecosystem.

A description of the near-surface ecosystems in future site investigations may contribute towards reducing the uncertainties. However, the possibility of a relatively quick evolution of the biosphere sets the ultimate limit on the accuracy of biosphere descriptions used to estimate radionuclide turnover far into the future.

Forestland probably has EDFs between those of agricultural land and peat bogs for most radionuclides. For this reason, classifying forest as peatland probably leads to an overestimation of potential doses.

The highest EDFs are obtained for the typical ecosystems peatland, well and agricultural land. In the well case, drinking water is the dominant exposure pathway for actinides. With regard to the more bioavailable chlorine, iodine and cesium isotopes, consumption of foodstuffs gives higher doses than consumption of water in the well case.

The probabilities of the different exposure pathways in the different ecosystems also need to be evaluated. Peatland, for example, requires several steps of human activity before high EDFs arise. The model for agricultural land, on the other hand, describes a more probable course of events, as does the calculation model for discharge in surface water.

Taken together, this means that the consequences estimated in SR 97 probably overestimate the risks considerably. With a better understanding of the dominant ecosystems and revised calculation models, most of the EDFs can probably be reduced.

Coming work

The calculation models for each typical ecosystem need to be reviewed in the light of e.g. new regulations. A model needs to be developed for forestland.

3.3.8 Calculations of radionuclide transport

Calculations of radionuclide transport constitute a large part of the safety assessment for a deep repository. The calculations are supposed to describe a large number of coupled processes in the repository, the surrounding rock and the biosphere. They require a large database and the body of results is large and complex. Calculations of radionuclide transport for the canister defect scenario are reported in this section. The purpose of the calculations is, in brief, to:

- describe radionuclide transport quantitatively for this scenario,
- illustrate the importance of uncertainties in input data and show which data have the greatest influence on the calculations results,
- compare the risk caused by the repository at the three sites with given acceptance criteria,
- illustrate the importance of the individual barriers in the repository system.

First the calculation models are presented, together with the confidence that they fulfil their purpose. Then a number of calculation cases are formulated with a view towards the above purposes. The calculation results are reported and the section is concluded with a discussion of results.

Description of the transport models

Radionuclide transport is calculated with the near-field model COMP23, the far-field transport model FARF31, and the dose model BIO42, see Figure 3-11. Input data to the calculations come from various more or less complex model calculations or data analyses of different conditions or phenomena.

The near-field model COMP23 is a so-called compartment model. COMP23 calculates how nuclides in a damaged canister are released from the fuel, how they may be precipitated due to solubility limitations, how the nuclides diffuse through the breach in the canister and on out through the buffer, and how they are transferred along different pathways to the flowing groundwater in the rock's fractures. Chain decay in the fuel and along all transport pathways is calculated.

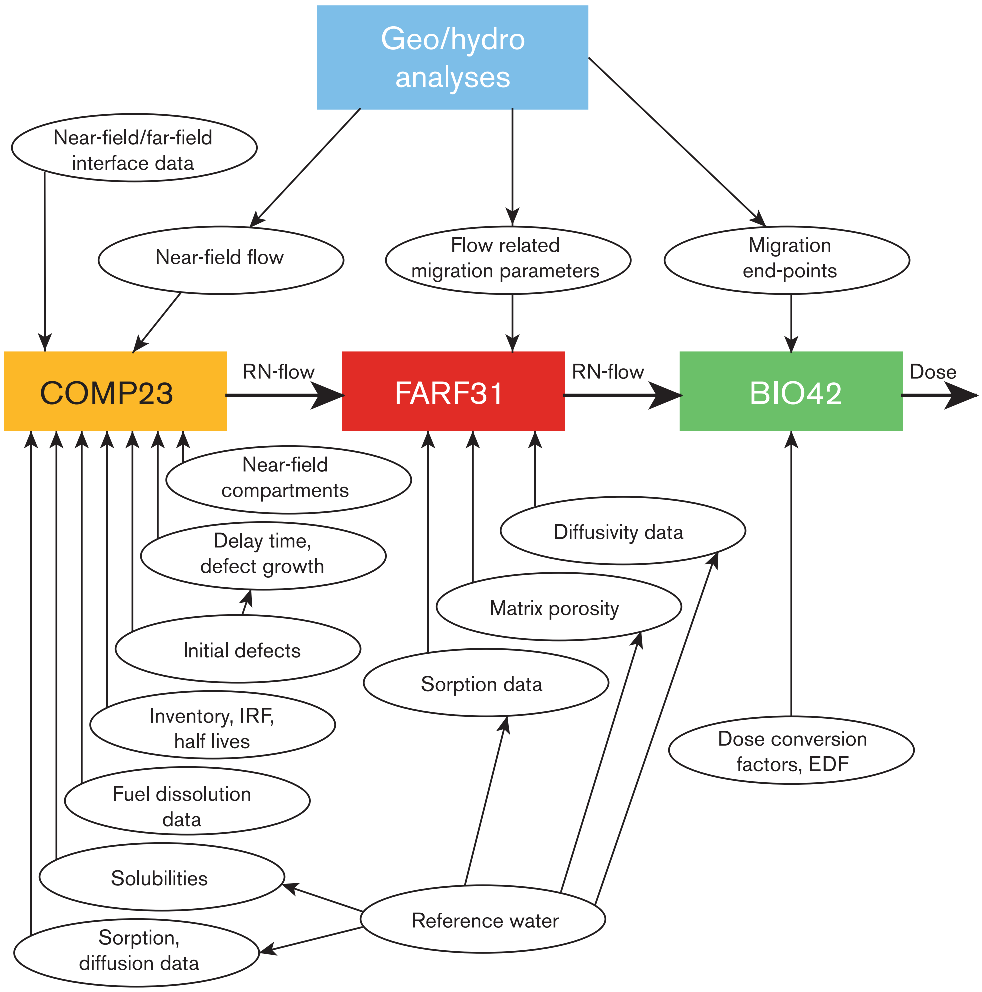

Figure 3-11. *Models for radionuclide transport.*

FARF31 is a so-called double-porosity model that is used to calculate transport of the released radionuclides through the rock. The model calculates radionuclide transport along the imagined flow path travelled by a fictitious particle following the groundwater flow through the rock. The model handles advection, dispersion, matrix diffusion, sorption in the rock matrix and chain decay.

The final dose calculation is done with the model BIO42, which calculates a dose by multiplying the release from FARF31 by an EDF value (Sv/Bq). The dose conversion factor is determined by what type of biosphere the release occurs to.

Confidence in the models for groundwater flow and transport

Confidence that the models in the safety assessment with the data used correctly describe relevant physical processes is important for confidence in the results of the model calculations. The models for groundwater flow and radionuclide transport are used directly to quantify the repository's safety in release calculations, and here the question of confidence is particularly important.

In the complete version of the main report, confidence in these models is discussed against the background of the ongoing international discussion of how the question of confidence should be handled.

The conclusion for the groundwater flow model HYDRASTAR is that confidence is good that the model correctly fulfils its purpose in SR 97. For the transport models COMP 23 and FARF31, confidence is judged to be good that the models certainly do not underestimate the consequences of the transport processes in the repository. Confidence that the calculated EDFs do not underestimate doses to man is also judged to be good, despite the fact that the mechanisms in some ecosystems are inadequately described.

What happens in the transport models?

The transport models are devised to ensure that the calculated releases will not be underestimated. The models therefore provide a simplified and pessimistic picture of the evolution. Here follows a brief description of the processes that are quantified in the transport models, as an introduction to the presentation of the results of the calculation cases.

1. No releases occur from the canister before a continuous water pathway has been formed between the fuel and the breach in the copper shell. Radioactive decay reduces the radionuclide content and total radiotoxicity of the fuel.
2. As soon as a continuous water pathway has formed, the instant release fraction of the inventory dissolves in the water. If the solubility limit is reached, the concentration of the dissolved nuclide in the water does not increase further. The nuclides dissolved in the water begin to diffuse out of the canister. The release rate of nuclides embedded in the fuel is determined by the rate of fuel dissolution. Here as well, the solubilities of the nuclides limit the concentration that can occur in the water.
3. The nuclides are sorbed with varying effectiveness in the buffer and the sorption processes determine the time for diffusion through the buffer. If this time is shorter than or comparable to the half-life of the nuclide, it continues out into the rock.
4. In the rock, the nuclide's sorption properties, together with the rock's transport properties (flow and fracture structure), determine the time for transport through the rock to the biosphere. In the same way as in the buffer, the half-life of the nuclide determines whether it passes through the geosphere before decaying.
5. In the biosphere, the nuclide gives rise to a dose that is dependent on its inherent radiotoxicity and how it is cycled in the biosphere type to which it is released. Both of these factors are included in the EDF.

In general, nuclides with a relatively high instant release fraction also tend to be readily soluble and relatively mobile in both buffer and rock. Several percent of the inventory of I-129, for example, is instantly released; iodine has very high solubility and is not sorbed in either buffer or rock. Plutonium isotopes, on the other hand, lie completely embedded in the fuel matrix, have low solubility and are sorbed strongly in both buffer and rock. Isotopes of uranium, thorium and americium have similar properties to plutonium. Furthermore, as a general rule the mobile nuclides are less radiotoxic.

Figure 3-12 illustrates schematically the fact that long-term radiotoxicity is dominated by those nuclides that have both low solubility and low mobility in buffer and rock.

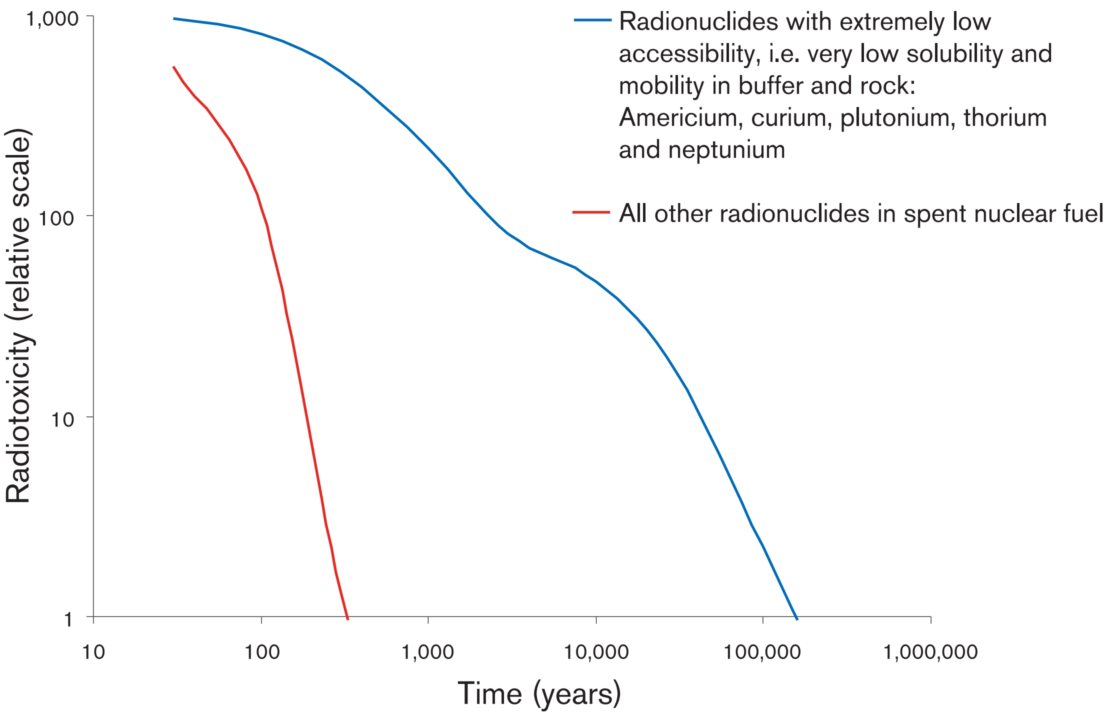

Figure 3-12. *Radiotoxicity of spent nuclear fuel broken down into nuclides with extremely low accessibility and other nuclides.*

The above description provides a general picture of the sequence of events and consequences. The quantitative details are determined by the input data chosen in a calculation case. The following account of different calculation cases is focused on the nuclides that make the greatest contribution to the releases. Many of the most toxic nuclides – such as isotopes of americium, plutonium and thorium – are not seen in the data, since the retarding capacity of the repository is very good for these nuclides.

Reasonable cases for Aberg, Beberg and Ceberg

Figure 3-13 shows the results from the calculations for Aberg, where all data have been chosen as reasonable. The calculation thereby applies to **one** canister, with an initial defect that has grown after 200,000 years so that a continuous water pathway to the fuel has been formed. Fuel dissolution and radionuclide transport thereby also start after 200,000 years. Many radionuclides have after this time completely decayed. Only very long-lived nuclides, mainly those with an instant release fraction, are released. The damaged canister is emplaced in a deposition hole with median values for groundwater flux and advective travel time.

The biosphere is a peat bog with reasonable EDFs. With today's flow situation at Aberg, most releases are expected to take place to the Baltic Sea. As a result of ongoing shoreline displacement, releases from several thousand years in the future can be expected to take place to land areas. The peat bog is the "land module" that is projected to give the highest doses, at the same time it cannot be regarded as improbable. This has therefore been chosen as the reasonable long-term case for Aberg.

The doses for all times lie far below doses from natural background radiation in Sweden. The dose curve is dominated by I-129, a long-lived, non-sorbing nuclide with a relatively high instant release fraction.

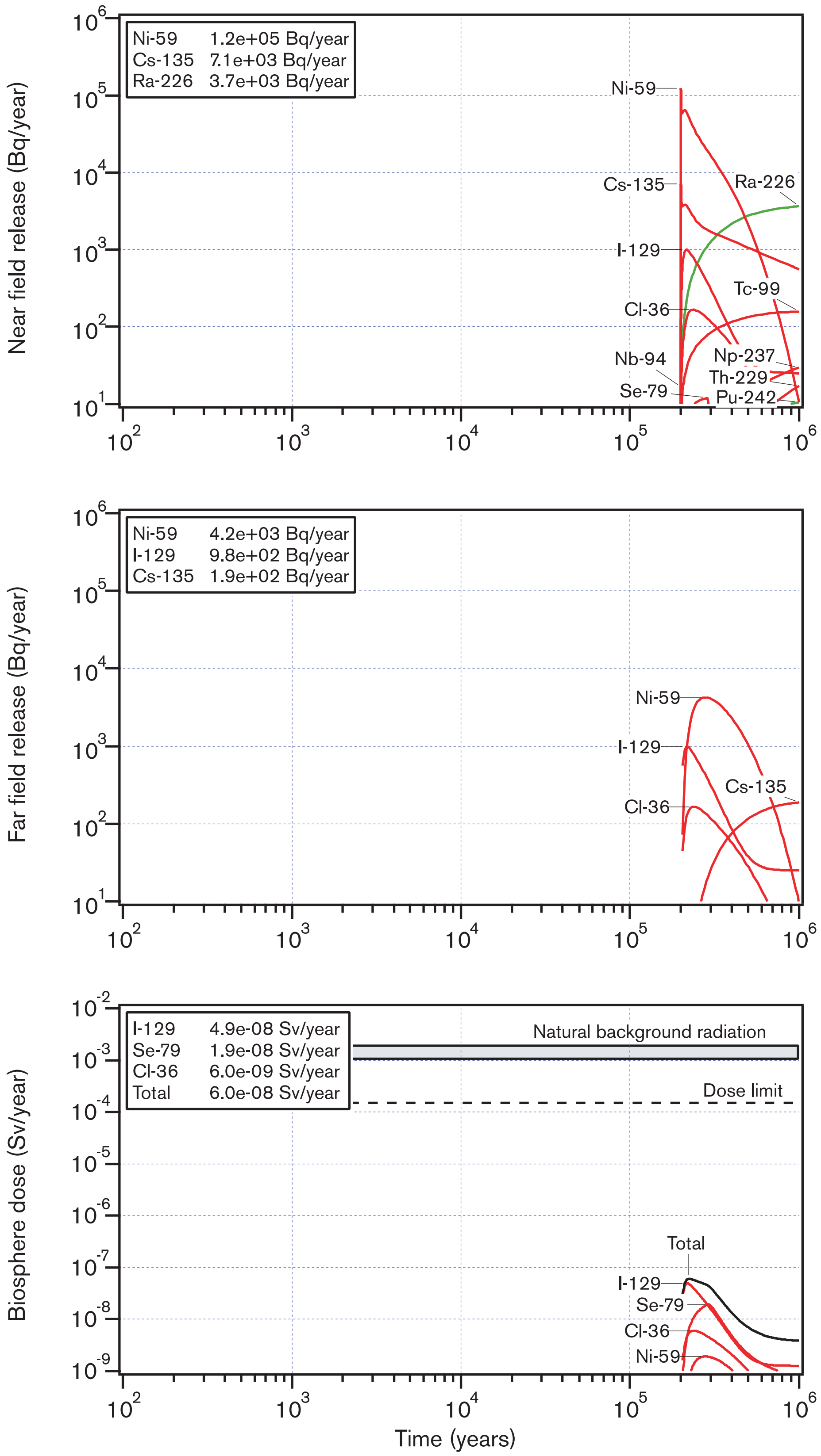

Figure 3-13. *Releases from near field and far field and doses in the biosphere as a function of time in Aberg. All data chosen as reasonable. The inset boxes show maximum releases and doses for dominant nuclides.*

The biosphere figures also show a dose limit of 0.15 mSv/y. This is the dose that corresponds to a risk of 10^{-5}/y if the exposure occurs with certainty. 10^{-5}/y is the risk limit that has been interpreted as being the relevant one to use for the well and peat modules in SR 97, see section 3.3.7.

The corresponding cases for Beberg and Ceberg are very similar, especially for the near-field releases. The geosphere's retention properties for e.g. Ni-59 and Cs-135 are better in Beberg and Ceberg. Here the geosphere releases are dominated by the highly mobile nuclides I-129 and Cl-36 where, as in Aberg, almost the entire near-field release makes it through to the geosphere. The dose curves (peat bog in Beberg and Ceberg as well) are similar on all sites, since I-129 dominates.

Uncertainty analysis

All input data to the radionuclide transport calculations are burdened with uncertainties. As described earlier, uncertainties have been quantified by determining not only reasonable values, but also pessimistic values for nearly all input data. Figure 3-14 is a compilation of how variations in each individual parameter affect the result on each of the sites. Regardless of which parameter is chosen pessimistically, the maximum dose lies below the dose limit.

The figure shows the change (increase) in maximum dose if the parameter in question is changed from a reasonable to a pessimistic value. The maximum dose pertains to the time up to one million years after deposition.

The figure shows that the greatest effects for Aberg are obtained for the number of initially defective canisters, the F factor (the product $t_w \cdot a_w$) and the dose factors in the biosphere (EDFs). Many of the other parameters have little influence on the result if they are varied, based on the thoroughly reasonable case. The effects of varying the F factor stem above all from poorer retention of Ra-226.

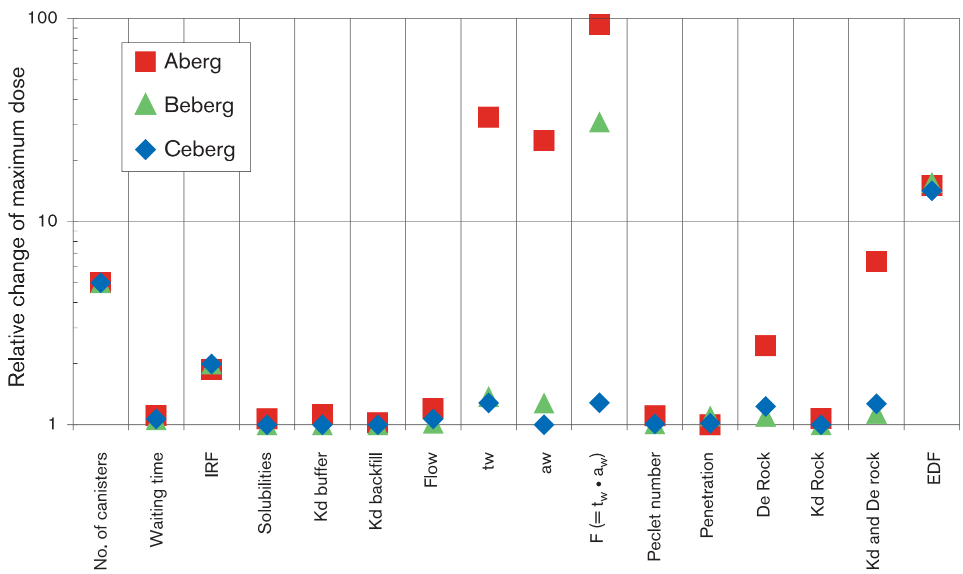

Figure 3-14. The effect of varying different parameters based on the thoroughly reasonable case.

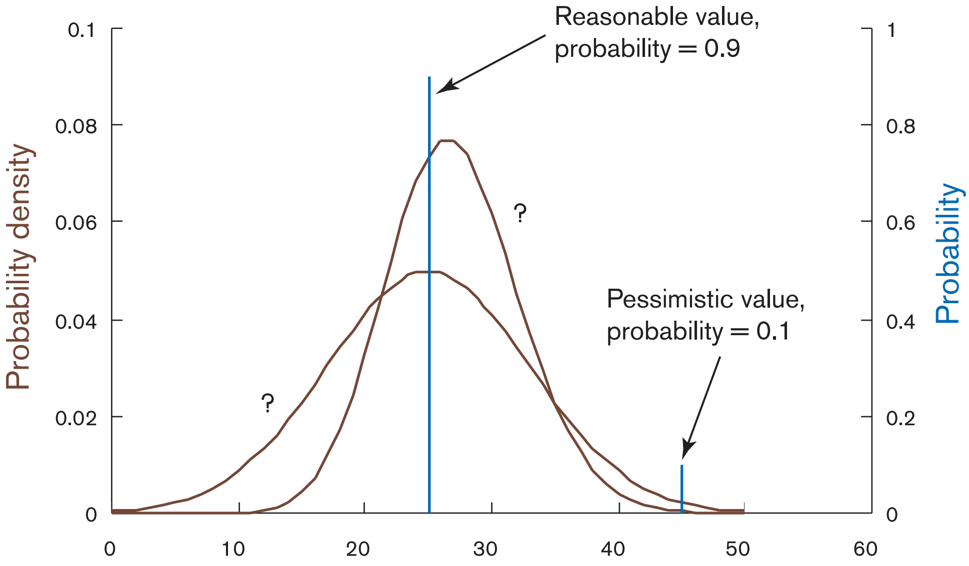

Figure 3-15. *Reasonable and pessimistic values, with probabilities of 0.9 and 0.1, respectively, are used for input data where distributions are lacking.*

Risk analyses

The stipulated criterion for repository safety is a risk measure, i.e. a summation of products of probabilities and consequences for different alternatives for repository evolution.

Calculations reported up to now show only the consequence in the form of dose for different choices of data for the models. To be able to calculate a risk, the probability of different data sets must also be estimated.

Probability distributions are only available for a limited portion of the underlying data, such as certain flow-related data in the geosphere. Only reasonable and pessimistic values are available for the large majority of data, with no estimate of associated probabilities.

The pessimistic values are chosen throughout to serve as an upper limit for the respective parameter's (negative) influence on the calculation result. Many pessimistic data must therefore be regarded as highly improbable. Reasonable data are also often cautiously chosen; in many cases, further research, more accurate measurements, etc. may lead to more positive data.

Based on the above, a probability of 0.9 is assigned to reasonable data and 0.1 to pessimistic data in the majority of cases where no distributions are available. The situation is illustrated in Figure 3-15. Instead of guessing what a whole distribution looks like, a rough estimate is made of the probability of reasonable versus pessimistic data. (The graduation of the x axis is arbitrary in this illustration; reasonable and pessimistic values can sometimes differ by several orders of magnitude.)

A probabilistic calculation is carried out for the three repository sites with data as follows:

- Calculated correlated distributions are used for advective travel times and fluxes at repository depth.
- Only pessimistic values are used for delay time, fracture geometries around deposition holes, Peclet number and maximum diffusion penetration depth in the rock matrix.

The reason is technical and related to limitations in current versions of the calculations programs. The influence of the difference between pessimistic and reasonable values on the calculation result is limited for these parameters.

- For remaining data, i.e. number of broken canisters, solubilities, IRFs, sorption data in bentonite, sorption data in backfill, diffusion and sorption data and flow-wetted surface area in the geosphere, and EDFs in the biosphere, probabilities of 0.9 and 0.1 are assigned to reasonable and pessimistic data, respectively.

Water flows at repository depth and advective travel times are correlated, according to the results of the hydrological calculations.

The results in the form of the dose's distribution function for the three sites are shown in Figure 3-16 and 3-17. The figures differ in that different biosphere modules have been used: a peat bog in Figure 3-16 and a site-specific well in Figure 3-17.

The reasonableness of the assigned probabilities can only be judged by studying the procedure used in the selection of reasonable and pessimistic values in each individual case, which is reported in the Data Report.

The results of interest for risk analyses are the mean values of the distributions. The mean value is a risk measure, since it represents a weighing-together of probability and consequence for all realizations. The mean values for the different distributions are therefore marked in the figures above.

The mean values should be compared with the risk limit in the figures. The risk limit is applicable to the most exposed individuals in a regional group, which is also true of the EDFs used for well and peat. The mean values for all sites lie well below the risk limit, in both the well and peat cases. It is also apparent that the safety margin differs between the sites: it is largest in Ceberg and smallest in Aberg.

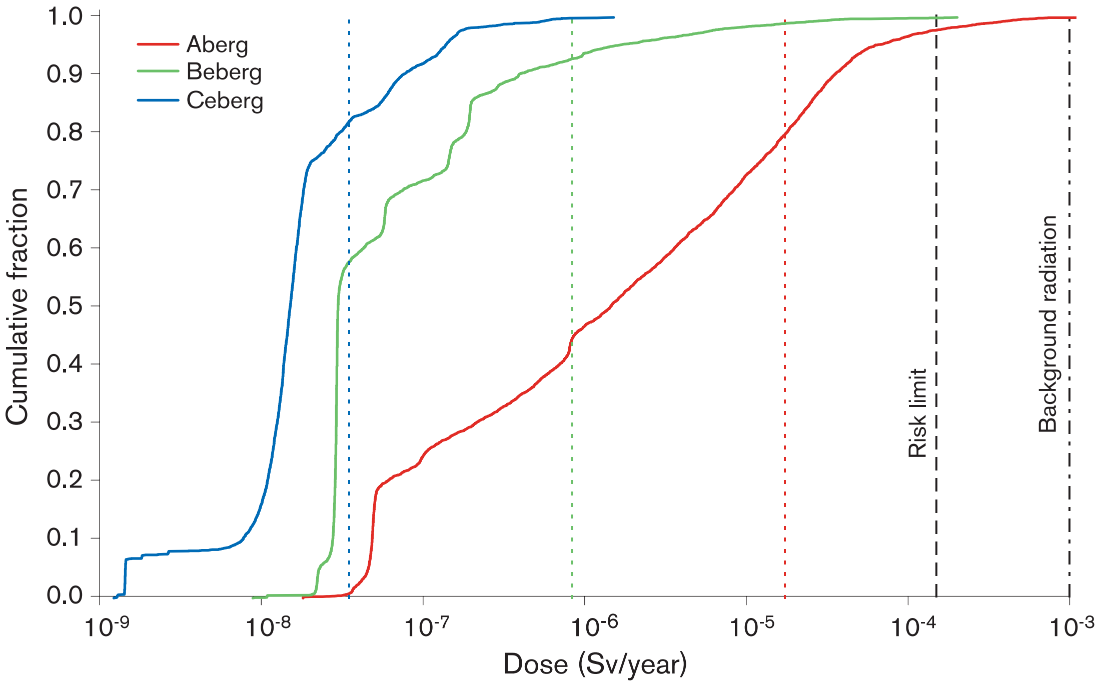

Figure 3-16. *Probabilistic results for site-specific peat bogs. The coloured vertical lines show the mean values of the different distributions, i.e. the calculated risk on the three sites.*

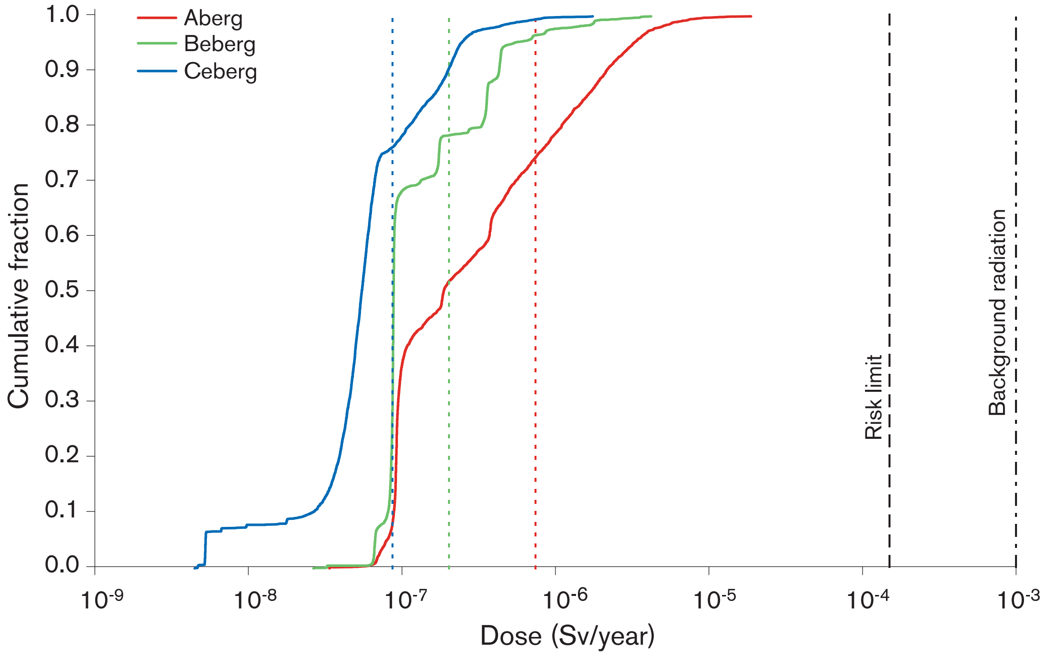

Figure 3-17. *Probabilistic results for site-specific wells. The coloured vertical lines show the mean values of the different distributions, i.e. the calculated risk on the three sites.*

It is also worth noting for the well case that **all** realizations for all sites are more than 10 times lower than the dose that corresponds to the risk limit in these calculations, i.e. the distributions in Figure 3-17 never come close to the risk limit.

Nearly all realizations that give significant maximum doses in the risk calculations are dominated either by I-129 or, if geosphere retention is weak, by Ra-226. A large canister defect is created after 20,000 years (pessimistic value in all realizations). The maximum dose of I-129 comes shortly thereafter.

The maximum dose of Ra-226 never arises until after more than 100,000 years, however, for fundamental reasons: Ra-226 is not present in the fuel initially, but is formed by chain decay of the initial content of U-236 and U-234. The nature of the decay chain is such that the rate of formation of Ra-226 does not become significant until after around 100,000 years.

In order to shed light on the importance of Ra-226, the result of the risk calculation can also be evaluated with Ra-226 excluded. The results are shown in Table 3-1. The results are influenced greatly by the peat ecosystem at Aberg and Beberg, in other cases more marginally.

The effect would have been roughly the same if the risk calculation had been done for a period of 100,000 years instead of a million years, since significant doses from Ra-226 come after roughly 100,000 years and for other nuclides that dominate the dose picture shortly after the canister breach has become large, i.e. 20,000 years.

In summary, the three sites are relatively equivalent during the initial 100,000 years. The differences between the sites are roughly a factor of 10 and the risks are always less than one percent of the risk limit, for both well and peat ecosystem. The same conclusion applies to the well case for the period between 100,000 and one million years. With a peat ecosystem, the risk increases at Aberg and Beberg by a factor of 10 and 20,

Table 3-1. Relative risks for peat ecosystem and well with and without Ra-226. The risks are expressed as fractions of the acceptance criterion for the most exposed individuals, i.e. 0.15 mSv/y.

	Peat ecosystem		Well	
	with Ra-226	without Ra-226	with Ra-226	without Ra-226
Aberg	0.11	0.003	0.005	0.001
Beberg	0.006	0.0006	0.001	0.001
Ceberg	0.0002	0.0002	0.0006	0.0006

respectively, but is always well below the risk limit. An important reason why this outcome is obtained for the peat ecosystem is accumulation of Ra-226 in peat, a process which is included in the biosphere model. To put the result in perspective, it can be noted that certain peat bogs in Sweden exhibit sharply elevated concentrations of Ra-226, which comes naturally from the bedrock.

The results should also be viewed in the light of the fact that extensive glaciations are to be expected in Sweden within a hundred thousand years, which is the subject of the climate scenario in section 3.4. A glaciation leads to erosion of virtually the entire soil layer. Aberg can be expected to be under the sea for a large part of the next hundred thousand years. These aspects are weighed together in the overall discussion of results of the entire assessment in the last chapter of the report.

Special cases

To shed light on the roles of different barriers in the canister defect scenario, a number of special cases have been calculated:

- the fuel is completely dissolved when a continuous water pathway is created,
- no solubility limitations,
- big initial canister defect,
- diffusion resistance in the buffer is neglected,
- retention in the geosphere is neglected.

None of the cases is realistic. They are included to illustrate the function of the barrier system. The changes that have been made in the models pertain only to radionuclide transport. The system's evolution in other respects, for example the hydromechanical consequences of a large initial canister defect, are not dealt with.

The point of departure in all calculation cases is reasonable data for Aberg. None of the special cases have consequences that exceed the dose limit, with the exception of the case where the solubility limits are excluded. The consequences of this case lie above the dose limit, but below the dose from natural background radiation. The special cases demonstrate that the multiple barrier principle works as far as radionuclide transport is concerned. Not even extreme and completely unrealistic assumptions concerning a single barrier function have unacceptable consequences.

Analytical calculations

In order to verify the results of the model calculations and demonstrate understanding of the transport models, some of the calculations of radionuclide transport in the near field and geosphere have also been carried out with simple analytical models.

Figure 3-18 shows emissions from the near field calculated with a simplified analytical model. Pessimistic values are used here for data related to canister damages and to the boundary layer between buffer and rock, and otherwise reasonable data for Aberg. Calculations of the same cases with the numerical model COMP23 show that the maximum releases differ by less than a factor of three for all nuclides in the figure.

An analytical expression for calculating how large a fraction of a release of a radionuclide passes through the geosphere without decaying is used for the geosphere. By using different combinations of the same reasonable and pessimistic data as in the model calculations, many of the conclusions regarding radionuclide transport in the geosphere for the three sites can be recreated.

3.3.9 Discussion of results

Risk analyses

The repositories at Aberg, Beberg and Ceberg meet the Swedish Radiation Protection Institute's acceptance criteria for a deep repository for spent nuclear fuel.

The conclusion is based on calculations with the cautious estimate that the probability that the pessimistic values will become reality is 0.1 throughout. An exhaustive discussion of the probabilities of reasonable versus pessimistic values is beyond the scope of this discussion. All data choices are thoroughly justified in the Data Report, and probability assessments must be based on the choices of data in the individual cases. The choice of a probability of 0.1 for pessimistic data is based on a general such evaluation.

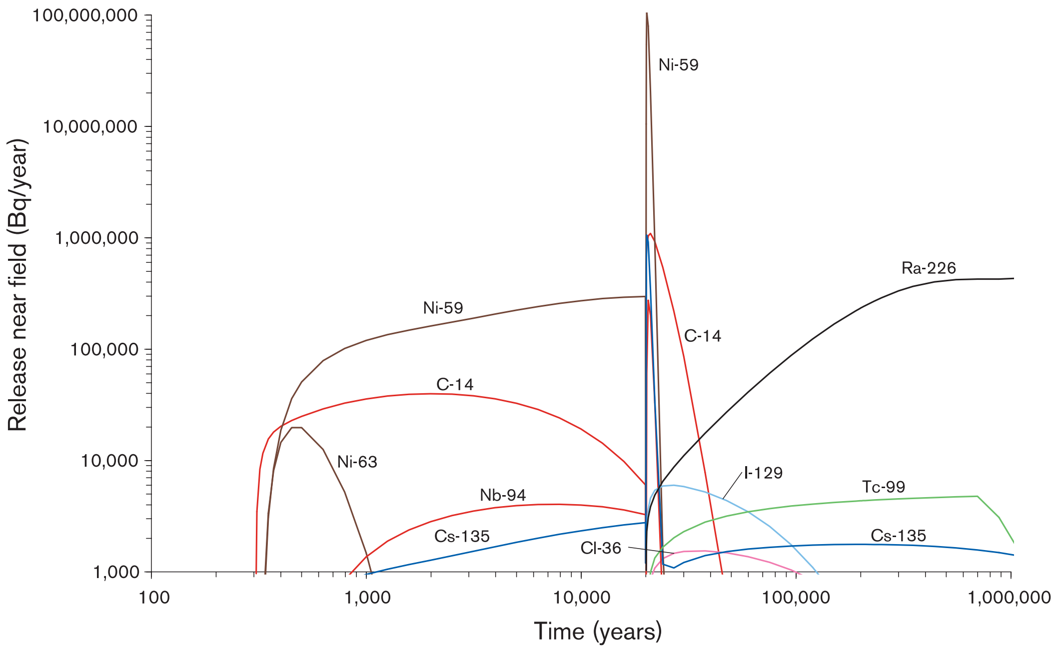

Figure 3-18. *Releases from the near field for pessimistic values for data related to canister damages and the boundary layer between buffer and rock at Aberg. Analytical calculation.*

If the release takes place to a well, the risk at Aberg is less than one hundredth of the acceptance criterion if the calculation is performed for a timespan of one million years. The risk at Beberg is approximately one-fifth, and at Ceberg one-tenth, of the risk at Aberg.

In the case of release to a peat ecosystem, the sites differ in roughly the same way for times up to around 100,000 years. For a million years, the risk at Aberg then increases by approximately a factor of 40, at Beberg by a factor of 10 and at Ceberg only marginally. The natural radionuclide Ra-226 dominates the consequences for the peat ecosystem for times over 100,000 years.

The calculated retention capacity is equivalent at the three sites for long-lived non-sorbing nuclides. For sorbing nuclides, retention is strongest at Ceberg and weakest at Aberg.

The risk calculations pertain to the maximum dose that arises in 100,000 versus one million years. The maximum dose always arises at times after 20,000 years. For times up to 20,000 years, even an initially defective canister provides considerable protection, even with the most pessimistic assessment of the course of events.

How do the sites differ?

Geosphere: The retention capacity of the rock is of importance for the repository's retarding function in the event of a canister defect. The capacity is nuclide-specific and is determined by the nuclide's half-life, nuclide-specific chemical factors and the flow properties in the rock. The latter are expressed by the F factor, i.e. the product of the advective travel time and the flow-wetted surface area.

From the discussion of uncertainties in the hydraulic modelling, it is also worth recapitulating the assessment that calculated differences in travel times and fluxes between the sites are probably to a large extent physically based and can be traced to site-specific hydrogeological conditions. The rock mass in Aberg and Beberg is, for example, approximately 100 times more permeable than the one in Ceberg, with the reservation that the investigation methodology also differs between the sites.

However, non-sorbing nuclides such as I-129 often dominate in the dose and risk calculations. They always penetrate through the rock, regardless of the F factor, so that the effect of differences in F factors between sites is evened out to some extent in a total risk analysis.

Biosphere: The biosphere conditions have a great influence on the calculated dose. The doses will be several orders of magnitude lower if the release takes place to watercourses, lakes or the sea, compared with if it takes place to land (peat, agricultural land, well). For Ra-226, which dominates the dose in many unfavourable realizations, the EDF for peatland is moreover more than two orders of magnitude higher than the well value.

Despite the great differences in EDFs, it is unreasonable to attach great importance to today's biosphere conditions in site discussions. The biosphere types peat, agricultural land and well exist or will probably exist on every conceivable deep repository site. The role of the biosphere in the analysis must also be regarded in the light of the extensive impact on the biosphere that follows from the expected climate changes.

Uncertainty analysis

The effect of substituting reasonable for pessimistic data is studied systematically in the uncertainty analyses. It is first observed that for the well case, the dose does not exceed the value equivalent to the risk limit in a single realization of the risk calculation at any site. The same applies to the peat module at Ceberg, while the dose has been exceeded in a few of the most unfavourable realizations at Aberg and Beberg if Ra-226 is included.

This is also a form of illustration of the multiple barrier principle: Reduced performance (pessimistic values) of one barrier does not lead to unacceptable performance of the system as a whole. The margins are greatest here for Ceberg and smallest for Aberg.

Canister: The difficult-to-assess uncertainties in the number of damaged canisters (reasonable value 1, pessimistic 5) directly affect the result. It is not apparent how these uncertainties can be reduced in the future.

The analysis of the canister's hydromechanical evolution after being damaged is surrounded by great uncertainties. Although reasonable and pessimistic estimations can be made today of the few factors that are used in the calculation of radionuclide transport, it is important to improve our understanding of the hydromechanical evolution.

Fuel: Another uncertain factor is fuel dissolution. The influence of uncertainties in fuel dissolution rate is limited for most nuclides.

Even a completely unrealistic assumption of immediate fuel dissolution has limited effects with otherwise reasonable data. The solubilities of the individual nuclides limit the release. The most important exception is Ra-226, for which the effect only becomes noticeable after more than 100,000 years.

It is important to create a better theoretical and experimental understanding of the fuel dissolution process.

Buffer: The effects of uncertainties surrounding the properties of the buffer as regards radionuclide transport are small. Provided that the buffer's long-term evolution is as in the base scenario, our understanding of the buffer's role in radionuclide transport is good.

Backfill: Uncertainties surrounding the influence of the backfill material on the total release of radionuclides from the near field are small. The conclusion must be verified against the results of more detailed analyses of the long-term evolution and general performance of the backfill material. The need for such analyses is pointed out in the discussion of results for the base scenario.

Geosphere: The simple uncertainty analyses show that the natural variability in the F factor in the geosphere has the greatest effects on the calculation result (maximum dose over 1 million years). Uncertainty in sorption and matrix diffusion properties are also important for retention, but are of less importance than the flow properties. The results point towards important conditions in the bedrock to determine in a site investigation.

The methods for determining the F factor can be improved. Experiments and simulations with alternative descriptions of the rock can improve the means for assessing this crucial parameter. This work will most probably show that the pessimistic values currently proposed are much too low.

The fracture structure and hydraulic properties around individual deposition holes are handled pessimistically in SR 97. Among other things, a small fracture has been assumed to intersect each deposition hole directly opposite the breach in the canister's copper shell. No deposition holes have been deselected as a result of the hydraulic calculations. On a real repository site with bored deposition holes, active choices based on observations in the deposition hole and calculations could improve the result of the analysis. By means of active choices on a small scale, large-scale differences between different sites could also be evened out. This would require a selection method that has been shown to work under realistic field conditions.

Biosphere: An important improvement of the data for the analysis ought to be able to be accomplished with an improved biosphere modelling. The conclusion is not really based on the factor of 10 that separates the reasonable values from the pessimistic ones, such differences are probably unavoidable. The assessment is rather that even the reasonable values are probably often greatly overestimated, justifying further work on the biosphere models.

The highest EDFs for many radionuclides are obtained for peatland and agricultural land, which is not in agreement with previous safety assessments where the migration pathway via well has been considered to make the dominant dose contribution. The capacities of peat and agricultural land to accumulate radionuclides are the basis of the high EDFs. The modelling of the EDFs in SR 97 is, however, simplified and probably overly pessimistic. It is also doubtful whether it is meaningful to calculate dose consequences hundreds of thousands of years into the future. Alternative safety indicators, such as activity releases, may be more meaningful since they make it possible to determine whether the repository will pose a general environmental problem or not.

The risk analysis is carried out using a method that permits probabilistic calculations, despite a limited body of statistics. It is urgent that the method be evaluated for coming assessments.

Environmental protection

According to the Swedish Radiation Protection Institute's regulations, the impact of the repository on surrounding ecosystems shall also be described. No specific methods for this are available today.

In most calculations cases in the canister defect scenario, the doses are many orders of magnitude below the natural background radiation. Based on this, the general assessment is made that the ecosystems at Aberg, Beberg and Ceberg are not adversely affected by the radionuclide migration in the canister defect scenario.

3.4 Climate scenario

The Swedish climate can be expected to change dramatically in a long time perspective. Biological and geological observations indicate large and cyclical climatic variations in the past. The cycles are closely associated with variations in the earth's motion around the sun and the cyclical variations in insolation to which this has given rise.

Variations in the earth's orbit are therefore expected to lead to future climatic variations that resemble the historic ones. Today's climate is warm by historical standards, and future changes are mainly expected to lead to a colder climate with extensive glaciations in a hundred-thousand-year perspective.

The climate is changing continuously over the long term in both time and space. In a general description of the climatic evolution, three primary climatic domains are spoken of:

- temperate/boreal domain,
- permafrost domain,
- glacial domain.

Temperate/boreal domain

This is today's climate in the greater part of Sweden. The climate is damp, which provides good conditions for vegetation and fauna and thereby also for agriculture and human settlement.

Even under temperate/boreal conditions, the country is influenced by former glaciations. The ice load depresses the earth's crust, which then strives to resume its original shape during ice-free periods. The process is slow in relation to the advance and retreat of the ice sheet, which means that the earth's crust is in constant motion. This is noticeable today above all in coastal areas, where the coastline is displaced when the land rises and is depressed (crustal upwarping/downwarping).

Permafrost domain

In this climate the ground, or at least parts of it, is permanently frozen but not ice-covered. Today, permafrost exists at high altitudes in northern Sweden. Since it can take thousands of years for permafrost to thaw, residues of permafrost from previous cold periods may remain during periods of warmer climate. Such permafrost occurs in Russia and Canada today.

The tundra ecosystem is characteristic in areas with continuous permafrost. The vegetation has adapted to the very short growing season and consists of herbaceous plants, thickets and shrubs.

The permafrost prevents infiltration and groundwater recharge. Most of the precipitation stays in the surface layer of the soil, and stream discharges vary widely over the year. The snow and ice melt during the span of a few weeks in the spring, causing peak runoff and stream discharges.

Conditions for agriculture and human settlement are much worse than in a temperate climate.

Glacial domain

The glacial domain is mainly distinguished by the fact that the ground is covered with ice over large areas. The soil under the ice cover is heavily eroded of organic material. The coastline is highly displaced, and areas in Sweden which are land today may be covered by sea during the glacial domain. Groundwater flows and groundwater pressures may be considerably changed compared with present-day conditions.

The water that infiltrates down to the bedrock during the glacial domain is expected to be oxygen-rich, since oxygen is not consumed by organic matter in the soil layer. The salinity of the groundwater varies mainly depending on whether the ground is covered by sea or ice. The ice also exerts a considerable mechanical load on the bedrock.

During much of the glacial domain, the prerequisites for agriculture and human settlement are non-existent.

3.4.1 Climate scenario in SR 97

A possible climatic evolution for Aberg, Beberg and Ceberg for the next 150,000 years is sketched in SR 97. The predictions are based on model simulations of the global climatic evolution based on expected variations in insolation (solar irradiance). Together with geological evidence, the global evolution and simulations of coastline displacement and ice dynamics provide a picture of the regional evolution in Scandinavia. Based on this, the site-specific evolution for Aberg, Beberg and Ceberg can be predicted. The evolution is described as a series of climate-driven domains that succeed each other, see Figure 3-19.

Site-specific thermal, hydraulic, mechanical and chemical conditions in the bedrock on the three sites are estimated from the climate descriptions.

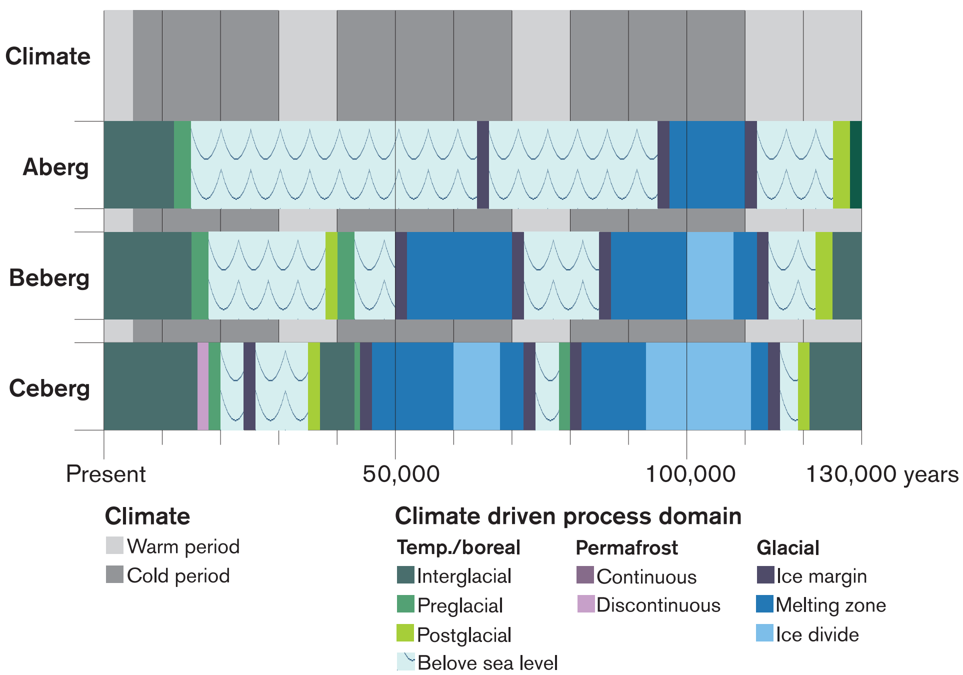

Figure 3-19. *Evolution at Aberg, Beberg and Ceberg as temporal series of climate-driven domains, and the time when the sites are covered by sea.*

By comparing these conditions with those that occur in the base scenario, the THMC evolution of the repository is predicted in the form of a comparison with the base scenario.

3.4.2 Impact on the repository

In relation to the base scenario, the transitions between colder climate, with permafrost and ice growth, and warmer climate, with conditions similar to today's, lead to the following changes:

- Conditions in the biosphere are radically altered.
- The temperature in the rock is affected.
- The conditions for groundwater flow are changed – the groundwater flow may both decrease and increase during different periods.
- The load conditions in the rock are altered during a glaciation.
- Altered groundwater composition – above all, the salinity of the water will vary.

The most dramatic climate-related changes occur in the biosphere. During a glacial/interglacial cycle, the biosphere is changed radically in several contexts, for example when land is transformed into seabed and vice versa, or when a continental ice sheet covers an area.

3.4.3 Thermal evolution

A colder climate lowers the temperature in the rock, but the permafrost is never expected to reach repository level. The impact on the temperature in canister, buffer and backfill will therefore be moderate. In parts of the geosphere where the groundwater freezes, groundwater flow in particular will be affected.

3.4.4 Hydraulic evolution

The groundwater flows change during a climate cycle, they can both increase and decrease relative to present-day flows. The groundwater flows at the three sites are expected to be less than or comparable to today's flows during most of the coming cycle. Besides by the fact that the groundwater freezes during a permafrost domain, the hydraulic evolution is greatly affected by the changes in hydraulic boundary conditions due to changes in precipitation, coastline displacement and ice growth in the surroundings. The groundwater pressure may increase considerably beneath an ice sheet, and the infiltration pattern may change. If a site becomes sea-covered, the pressure differentials in the geosphere will be evened out and the hydraulic conditions may become virtually stagnant.

3.4.5 Mechanical evolution

Mechanically, the boundary conditions are changed radically by the weight of an ice sheet. Both the static load from the ice sheet and the dynamic processes when the ice sheet advances and recedes have mechanical repercussions in the repository rock. The loading situation is complex: ice thickness, water pressure, crustal downwarping, large-scale tectonic movements and the properties of the rock mass and the fractures affect the state of stress. The interaction between the mechanical load and the high groundwater pressure can cause fractures to both widen – hydraulic fracturing – and close, thereby affecting the groundwater flow. In other words, feedback is obtained to the hydraulic evolution. Earthquakes induced by ice loads are dealt with in the earthquake scenario.

Fracturing and/or propagation of existing fractures can occur if the stresses and stress anisotropy generated by the ice load are sufficiently great. Increased principal stress differences can lead to shear failure in the rock mass. In the analyses done to date, however, the additional stress from the ice load has only lead to increased stress anisotropy in the vicinity of the tunnels and the deposition holes.

Glaciations are therefore not expected to lead to the formation of new large fractures that intersect deposition holes, or to the propagation and fusing of existing smaller fractures so that such large fractures are formed.

The maximum hydrostatic pressure at repository depth is estimated in the climate scenario to be about 32 MPa. With the swelling pressure added, the isostatic pressure on the canister could thereby approach 40 MPa. The canister insert in the BWR design has been calculated to withstand an evenly distributed external pressure of 80 MPa, while in the PWR design it withstands 110 MPa, see section 3.2.6. This means that the canister should withstand the pressure increases that can occur during a glaciation.

The calculations of canister strength need to be refined with more realistic, inhomogeneous material properties as per section 3.2.6.

3.4.6 Chemical evolution

Chemically, the boundary conditions are changed compared with the base scenario by the fact that oxygen-rich water infiltrates down into the bedrock without the oxygen being consumed in an organic soil layer, since this layer is expected to be obliterated during large parts of a glacial cycle. The oxygen is instead consumed mainly by reactions with the minerals in the geosphere.

The conclusion of studies of the transport of oxygen-rich glacial meltwater through the bedrock is that the rock has the capacity to consume the oxygen before it reaches repository level. At points where the gradients for groundwater flows are temporarily highest and hydraulic conductivity is great, however, oxygenated water could be carried down to great depths. This means that if oxygenated water was to exist at repository level, its occurrence will be limited, both in time and space. In the unlikely event that oxygenated water should reach a deposition hole, the bentonite buffer has a high capacity to consume oxygen, so oxygenated water should never reach the canister.

Climate change also causes changes in the salinity of the groundwater. In the first place, salt is excluded from the water when permafrost forms so that the still-unfrozen water has a higher salinity. In the second place, the altered flow conditions in the geosphere create new conditions for transport and mixing of water from different parts of the geosphere and infiltrating surface water. The result can be both higher and lower salinities than in the base scenario.

During the temperate/boreal domain the expected salinity variations lie within the variations between non-saline and saline water that exist today on the three repository sites. When the ice front passes, high salinities can occur at repository depth. Higher salinity results in faster ion exchange and thereby faster lowering of the swelling pressure to 4–5 MPa, which is reached after completed ion exchange from Na^+ to Ca^{2+}, see the base scenario. The function of the buffer would not be jeopardized even if the high salinities that might conceivably occur briefly when an ice front passes were permanent, according to the account in the base scenario.

The low salinities that can temporarily occur at repository depth on a site that is located close to the ice margin lie slightly below the range reported in the base scenario. Lower salinity results in slower ion exchange. The estimated concentration of divalent cations is sufficient to prevent buffer erosion, but the margin is small.

In summary, the expected changes in groundwater composition during a glacial cycle are judged not to threaten the integrity of the engineered barriers.

As in the base scenario, the conclusion is drawn that the long-range effects of erosion under extreme conditions may possibly need to be further studied. The chemical evolution and function of the backfill in the climate scenario remains to be analyzed.

3.4.7 Radionuclide transport

Radionuclide transport only occurs if there are leaking canisters in the repository. The changes during a glaciation are not expected to cause canister damage resulting in loss of integrity. The conditions for groundwater flow, and thereby also the groundwater composition, change during a glaciation. The changes affect the transport of radionuclides from the repository and through the rock, providing there are initially leaking canisters. The conditions in the biosphere are altered radically during a glacial/interglacial cycle. This greatly influences the doses to which man may be exposed.

During periods with high groundwater flows, the potential outward transport of radionuclides from a repository with leaking canisters increases. Increased groundwater flows can be expected in coastal locations undergoing land uplift, and above all during certain phases of a glaciation. When the groundwater flows are at a maximum, water turnover at the surface is also great. This means that the dilution of radionuclides in the biosphere is expected to increase. These factors reduce doses relative to present-day conditions.

A relative increase of doses is only expected at Aberg when the relative coastline falls compared with the current level so that sea-covered areas become dry land. This only occurs during interglacial periods during the transition to a colder climate, which means on the order of every hundred thousand year. The increased doses are above all due to changes in the biosphere. The importance of this change is studied in the canister defect scenario, where the effects of postglacial land uplift at Aberg are included in the reasonable case.

3.4.8 Summary

The climate scenario describes a postulated evolution during the coming glacial/interglacial cycle. It is based partly on modellings of the evolution during this period, and partly on the pattern that has characterized the past 400,000 to 900,000 years. The sketched evolution can therefore be seen both as a scenario for the coming 130,000 years and a general description of the course of events during an arbitrary glacial/interglacial cycle.

The most dramatic climate-related changes occur in the biosphere. During a glacial/interglacial cycle, the biosphere is changed radically in several contexts, for example when land is transformed into seabed and vice versa, or when a continental ice sheet covers an area. The load from a continental ice sheet influences the mechanical evolution. The composition of the water that is carried down in the rock, the boundary conditions for the groundwater flow through the rock, and thereby the groundwater composition will be altered during a glacial/interglacial cycle. The changes that are expected are not judged to jeopardize the integrity of the engineered barriers. Climate-related changes are not expected to affect canisters resulting in loss of integrity.

3.4.9 Uncertainties

It is very probable that the climate in the future will become colder and that Scandinavia will at some time be covered with ice. It is likely that such a situation will occur sometime during the next 100,000 years. However, our knowledge of the earth's climate system is not sufficient to make any exact predictions of the extent of the ice cover during the coming glacial/interglacial cycle.

The climate scenario in SR 97 describes a course of events that resembles the one we recognize from the most recent glacial period. The scenario contains three cold periods and three warm periods. During the last cold period, the extent of the ice cover is assumed to correspond to that we know from geological data. During the last warm period, interglacial conditions are assumed. The intention is to cover changes of importance for the evolution of the repository in such a way that their consequences are not underestimated.

Confidence in coverage of possible changes is therefore good, even if the descriptions of details in the climatic evolution are burdened with great uncertainties.

As far as the evolution of the repository system under the changed conditions is concerned, uncertainties exist with regard to:

- Canister strength, where calculations need to be refined by the use of more realistic material properties.
- Buffer erosion with extremely ion-poor groundwater compositions.
- Evolution and function of the backfill in connection with climate change.

3.5 Earthquake scenario

3.5.1 Background

The earthquake scenario examines how earthquakes can affect repository safety. The analysis is centred on the question of whether earthquakes can lead to a breach in isolation in one or more canisters. The premises for the scenario are the same as for the base scenario, except that earthquakes are assumed to occur in the surrounding area. Otherwise, as in the base scenario, the repository is built according to specifications, and present-day conditions in the surroundings are projected to persist.

Present-day tectonic conditions in the Baltic Shield on which Sweden rests are judged to have been constant over the past 2 million years, while the state of stress may in all essential respects have endured for the past 25 million years.

The Shield has also been subjected to cyclical loadings due to glaciations, and will probably be subjected to new such loadings. The timescale for a glaciation cycle, approximately 100,000 years, is short in the tectonic time perspective, and it can be assumed that the state of stress is largely restored between the glaciations. There are, however, questions with regard to what state of stress will prevail under the ice cover and at its margin, and what timescale applies to the recovery of the stress field.

The large-scale movements that take place in the Shield determine the boundary conditions for the long-term mechanical evolution of the host rock.

Earthquakes

The Swedish bedrock is of ancient origin and exhibits, as do other shield areas, a low level of seismic activity in relation to tectonically active areas. In the Swedish part of the shield, earthquakes mainly occur in the southwest in the Lake Vänern area and along the Norrland coast. Only on a few occasions has the magnitude of the quakes exceeded 4.0. Quakes that occur in one of the earth's tectonically active regions – such as Japan, the Caucasus or California – may have a magnitude of around 8, which means that nearly a million times more energy is released than by a quake of magnitude 4.

An earthquake represents a sudden release of strain energy that has accumulated during a slow process of deformation, generally via shear movements along a major or minor fracture zone. The crystalline basement in Sweden has been deformed in many periods and has well-developed systems of fractures and fracture zones. Future earthquakes will therefore most probably be triggered in existing fracture zones. Quakes of great magnitude can then be expected to take place in large zones, since they constitute the weakest links in the bedrock, and since their large extent in the horizontal plane permits large movements. Statistics from earthquakes all over the earth show a clear connection between the magnitude of the quakes and the extent of the zones in which the quakes occur.

There are two different opinions regarding what type of large-scale deformation is the main cause of the current seismic activity in Scandinavia:

1. Deformations caused by plate-tectonic movements.
2. Deformations caused by differential land uplift resulting from postglacial rebound.

Earthquakes with much greater magnitudes than we know from historic time occurred during the most recent deglaciation period. Earthquakes with a magnitude of approximately 8 occurred then in northern Sweden. Similar quakes may also have occurred in other parts of Sweden during this time. The cause of these large quake has not been established, but the strong coupling to a limited period of time during the retreat of the continental ice sheet puts the focus on a rapid differential land uplift process or constrained tectonic stresses during the long ice-covered period as possible causes.

Canister

Based on model calculations for the canister reported in section 3.2.6, the pessimistic criterion that rock movements of a magnitude of 0.1 metre and more around a deposition hole can lead to canister damage is used in the following earthquake analysis.

3.5.2 Analysis of earthquake risks

The seismic activity in the Swedish bedrock today shows that some of the strain energy that is continuously built up or has been built up during some earlier era is released by earthquakes. The risks to the safety of the repository that can arise due to the fact that the mechanical effects of quakes propagate into the repository area must therefore be analyzed and evaluated.

In order to estimate the number of canisters that can be damaged due to quakes within the next 100,000 years at the three repository sites, it is necessary to calculate or estimate at how many deposition holes isolated or cumulative displacements along fractures may exceed 0.1 m during the period. Such calculations have been carried out for the three sites Aberg, Beberg and Ceberg. The analysis includes:

A. Simulation of the mechanical effect on the rock mass of individual quakes of different magnitudes and at different distances from the repository.

B. Prediction of how the frequency of quakes of different magnitudes varies in time and space during the next 100,000 years.

C. Site-specific simulations of individual quakes according to the magnitude/frequency statistics that apply in the area in question with site-specific occurrences of fractures and fracture zones.

A. Effect of single quakes

The most important parameters that control a secondary fracture movement, e.g. at a deposition hole, are the magnitude of the quake, the distance from the fracture zone where the quake occurs to the fracture that intersects the deposition hole, and the size of the fracture, i.e. its extent in its own plane. Generic examples show that fracture movements of 0.1 m can be reached if a quake of magnitude 7.5 occurs within a distance of 100 m from the repository. In a similar manner, fracture movements of 0.1 m can be reached if a quake of magnitude 8.2 occurs within a distance of approximately 1 km from the repository. The examples are calculated with the same pessimistic assumptions concerning the properties of the rock that are discussed in section 3.5.3 below. Above all, friction in the fractures is neglected.

B. Prediction of earthquake frequency

The strategy is to use existing earthquake statistics and extrapolate them 100,000 years into the future. The statistics covering the magnitude and frequency of Swedish quakes only include quakes with magnitudes of less than 5. The existing statistics are extrapolated to apply to quakes with magnitudes up to and including 8.5. The statistics are not extensive enough to allow any certain conclusions to be drawn regarding differences between the three repository sites.

C. Site-specific simulations of earthquakes

The simulations are done in four steps:

1. Fracture data from the three repository sites are analyzed and used to devise fracture network models. The fracture network models have a deterministic portion and a stochastic portion.

2. For each repository site, 100 realizations of the stochastic fracture network model are computer-generated.

3. For each repository site, earthquakes are randomly distributed among zones situated within a distance of 100 km from the repository site in accordance with the earthquake statistics that are assumed to apply to the area. Regression relationships from earthquake statistics are used to determine what minimum extent is required to accommodate the quake. A single quake is represented as an instantaneous displacement.
4. The effect of the quake on a statistically representative selection of canisterintersecting fractures is analyzed by means of computer simulation.

Results

The results of the analysis are expressed as the percentage of deposition holes that are subjected to fracture movements greater than 0.1 m in 100,000 years. The results are equivalent for the three sites, and the mean values for the number of damaged canisters are fractions of a percent.

The canister failure distributions are strongly asymmetrical. The median value for all sites is zero canister failures. In approximately 90 percent of the simulations, no canister failures at all occur for most combinations of sites and earthquake statistics. In the remaining 10 percent, single quakes usually cause the failures; the cumulative effects are thus small.

3.5.3 Uncertainties

The analysis is surrounded by many uncertainties, both ones that are handled pessimistically and others.

Important uncertainties that are handled pessimistically:

- **Mechanical properties of fractures.** The fractures are assumed to be without friction and cohesion. This is very pessimistic; if the fractures had had a strength, all secondary displacements would have been smaller. A friction angle of 30 degrees would have reduced the displacement sums by a something like a factor of 5. This would lead to no canister damages at all.
- **Mechanical properties of the rock mass.** The fractures' secondary displacements are not affected in the simulations by other fractures of by the cavities in the repository. The rock is assumed to behave in a linear-elastic manner. This is also pessimistic.
- **Canister positions.** There is uncertainty as to how successful actual efforts will be to avoid depositing canisters in holes intersected by large fractures. In the analysis, no credit has been taken for the fact that it will be possible to reject unsuitable canister positions.
- **Criterion for canister damage.** The criterion is pessimistic and is based on a weaker canister design that gave a moderate plastic strain in the canister at this deformation. The relevance of the criterion ought to be tested by new calculations with the current canister design.

The most important uncertainty that is not handled pessimistically is related to **extrapolation of the earthquake statistics**. The existing statistics cover only a limited time interval and no quakes of a magnitude greater than 5. Furthermore, there is no clear-cut answer to the question of the causes of former and ongoing seismic activities or the possible scope and duration of future glaciations.

3.5.4 Conclusions

The analysis methodology is the result of a first stage in the development of a procedure for quantitative analysis of earthquake scenarios.

Even today's pessimistic analyses show that the probability of canister damages during a period of 100,000 years is of the same order of magnitude as the one assumed for initial canister damages, i.e. a fraction of a percent is affected.

The pessimistic assumptions that are made in the risk analysis are judged with good margin to be able to compensate for the uncertainties that are associated with the prediction of future earthquakes. Just by making more realistic assumptions about the mechanical properties of the fractures, for example, such a great reduction of the movements is obtained that no canister damages are expected, provided that the repository is not located closer than 100 m to zones with an extent of more than 100 km. Such zones can with great certainty be avoided in the positioning of a future repository, which makes the risk of canister damages caused by earthquakes negligible.

Added to this is the effect of other pessimistic assumptions and pessimistically handled uncertainties, where especially the assumption that the rock mass is a linear-elastic medium without damping characteristics leads to large margins.

In the light of these considerations, earthquakes are not expected to lead to canister damages. For this reason, no calculations of radionuclide transport are performed in the earthquake scenario in SR 97. The methodology for earthquake analyses is being refined for future safety assessments.

3.6 Intrusion scenario

Concentration and isolation of the spent nuclear fuel on a site means that this site contains more hazardous material than other sites with similar natural conditions. As far as future use of the site is concerned, this inevitably means that certain restrictions will apply to people in the future as well. There is nothing unique about this in our society. There are many examples of sites of human activity – cities, mines, harbours, water sources, refuse tips, arable fields, rock tunnels, etc. – that have a shorter or longer impact on possible/permissible uses of the original natural site. The restrictions on a deep repository site ought to be minimal in view of the long timescale. In principle, it should be possible to use the site for anything.

The principle of collecting hazardous waste in one place poses a risk that people will be exposed to a large quantity of the waste. If future generations forget the repository and its purpose, they may inadvertently impact it. At worst, this human impact can lead to a breach in the isolation of the waste.

Human impact has been taken into consideration in the design of the repository and in the site selection process. The depth of the repository should allow human activity on and in ground close to the surface of the earth at the repository, as well as construction of many of the kinds of rock facilities that exist today. The site is free of what might be regarded as natural resources today.

However, due to uncertainties regarding the future development of human society, it is not possible either to guarantee that there will be no human impact on the repository or to describe what the nature of such impact will be. Large rock works near the repository may, in combination with natural processes, affect the performance of the engineered barriers. If a borehole is drilled through a canister, its isolation will be breached immediately.

As an illustration, the consequences of inadvertent penetration of a canister by drilling have been studied. Dose and risk to the drilling personnel, as well as to a family that settles on the site at a later point in time, have been investigated. The drilling personnel are exposed to the highest doses. If the hole is drilled within about 300 years after repository closure, the dose to the drilling personnel may significantly exceed the background radiation and reach the levels that are permitted today for people in radiological occupations. Since the probability of drilling through a canister is low – it is estimated at 10^{-7} – the risk that drilling personnel may be exposed to in the future because of the construction of the deep repository will nevertheless be very small, even if the repository should be forgotten. The dose to the family will be lower than to the drilling personnel, it will never exceed the natural background radiation. If it is assumed that there is an abandoned borehole through a canister on the site when a family moves there, the dose to the family may exceed the acceptance criterion of $1.5 \cdot 10^{-5}$ Sv per year up to 400 years after repository closure. Based on the estimated probability of drilling through a canister, at least 25,000 boreholes per year to a depth of 500 metres would be needed for the risk to amount to 10^{-6} per year.

The review of human activities that could affect repository performance shows that drilling of deep holes, and construction of large rock facilities, might harm the repository. It is possible that the restrictions on such use of the site will at some time be forgotten or violated. The question is then what consequences can be allowed. In the analyzed scenario, the rock investigators proceed carelessly – more carelessly than can be expected of people today with access to present-day technology. However, they do bear some responsibility for their actions.

In a long perspective, the repository site may be utilized in such a way that the repository is impacted. The possibilities of assessing the radiological consequences of this are limited, due to uncertainties associated with the development of human society. In the analyzed scenario, the most exposed individuals are exposed to a radiation dose equivalent to about 0.1–10 times the natural background radiation. The probability of such exposure is judged to be very small, making the risk to individuals (both drillers and those who may settle on the site after drilling) much less than 10^{-6} per year.

4 Discussion and conclusions

The purpose of SR 97 is formulated in the introduction in four points:

1. SR 97 shall furnish supporting data to demonstrate the feasibility of finding a site in Swedish bedrock where the KBS-3 method for deep disposal of spent nuclear fuel meets the requirements on long-term safety and radiation protection that are defined in SSI's and SKI's regulations.
2. SR 97 shall demonstrate a methodology for safety assessment.
3. SR 97 shall furnish supporting data to specify the factors that serve as a basis for the selection of areas for site investigations and derive which parameters need to be determined and which other requirements ought to be made on a site investigation.
4. SR 97 shall furnish supporting data to derive preliminary functional requirements on the canister and the other barriers.

Each of the points is discussed in this chapter, followed by an account of how experience gained from SR 97 can be used to prioritize research efforts. The chapter is concluded with a few short words that put SR 97 in its context within SKB's siting programme.

4.1 Safety of KBS-3 method in Swedish bedrock

The point of departure of the safety assessment is the post-closure state of the repository system, and the assessment analyzes how the repository changes with time. Three fictitious repositories based on data from three actual sites are analyzed to shed light on various conditions in Swedish granitic bedrock. Long-term changes are analyzed by a classification into internal processes in the repository system and external forces exerted by the environment. Different conditions initially and in the surroundings yield a set of scenarios for which the evolution of the repository is analyzed quantitatively. An assessment of the results of the analysis embraces both qualitative and quantitative questions:

- Are all internal processes and external events of importance identified?
- What are the results of the different scenario analyses and what confidence can be attached to the results?
- How should the results of the scenario analyses be weighed together into a total risk analysis?
- How do different conditions in Swedish bedrock affect the feasibility of building a safe repository?

The first three questions emerge from the methodology for the safety assessment, the last concerns a specific purpose of SR 97.

4.1.1 Are all internal processes and external events of importance identified?

Identification of internal processes and external events of importance for the long-term evolution and safety of the repository has proceeded for several decades, in Sweden and in other countries. The work is documented in numerous reports and databases. SKB has its own database, which is constantly maintained and updated with international results. Very few new processes of importance have been identified in the past decade. The structure of the repository with multiple barriers with partially redundant functions also reduces its vulnerability to a possible unidentified process.

The choice of scenarios in SR 97 is an expert judgement, based on experience from previous assessments, available databases, etc. A comparison with safety assessments in other countries shows that the set of scenarios that is analyzed in SR 97 agrees very well with other assessments.

Confidence that all important points have been included must be judged on the basis of the efforts that have been made to achieve completeness and expert knowledge of the repository system and its natural surroundings.

SKB judges that the internal processes and the set of scenarios that are analyzed in SR 97 are sufficient to provide good confidence in the results of the safety assessment.

4.1.2 What are the results of the different scenario analyses and what confidence can be attached to the results?

The following scenarios are analyzed in SR 97:

- A base scenario where the repository is conceived to be built according to specifications, where no canisters have initial defects and where present-day conditions in the surroundings are assumed to persist.
- A canister defect scenario which differs from the base scenario in that a few canisters are assumed to have initial defects.
- A climate scenario that deals with future climate change.
- An earthquake scenario.
- A scenario that deals with future human actions that could conceivably affect the deep repository.

Base scenario

In the base scenario, the evolution of the repository's isolating function is analyzed under the assumption that the present-day climate persists. The overall conclusion is that the canister retains its isolating capacity, even in a million-year perspective. This means that no releases of radioactive substances occur from the repository. The long-term function of the buffer and the long-term stability of the bedrock are important questions in the base scenario. The consequences of all known thermal, hydraulic, mechanical and chemical processes of importance are analyzed for both. Model studies and pessimistic rough calculations show that the buffer can be expected to retain its function in a million-year perspective. In the same way, it is shown that the geosphere, and particularly the rock volume used for deposition, can be expected to remain stable over a very long time.

The evolution of the buffer and geosphere provide premises for quantifying the canister's thermal, hydraulic, mechanical and chemical environment. Based on this, canister isolation is projected to be retained for a very long time. The calculated safety margins against both mechanical and chemical stresses in the base scenario are large even in a million-year perspective.

The analysis assumes that the temperature on the canister surface does not exceed 100ºC and that the water at repository depth is oxygen-free. The former premise can always be ensured by a suitable deployment of deposition holes. Extensive sampling indicates oxygen-free conditions in the groundwater at depths greater than a hundred metres. The fact that oxygen-free conditions prevail at repository depth is as a rule ensured by biological processes in the soil layer in connection with groundwater recharge; furthermore, there is a very great potential for oxygen consumption in the minerals in the geosphere.

Our fundamental understanding of the processes involved and confidence in models and data are discussed systematically in the Process Report and in conjunction with the specific analyses in the base scenario. Process understanding and confidence can generally be said to be good. Most of the processes are well-known and have been studied by scientists for many decades. Models and data are sufficiently reliable for the often rough estimates that are required to set pessimistic bounds on the evolution in the base scenario.

The results of the base scenario are in part a consequence of the safety principles that have served as a basis for repository design. Copper is stable in the repository's oxygen-free environment. The buffer consists of a natural clay taken from a geological setting where it has been stable for millions of years. The Swedish bedrock has been stable for an even longer time.

Canister defect scenario

In the canister defect scenario, the course of events in an initially defective canister and the radionuclide migration in buffer, geosphere and biosphere that can result are analyzed.

The overall conclusion is that the repositories at Aberg, Beberg and Ceberg meet the acceptance criteria for a deep repository. The risk calculations are designed to ensure that the risk is not underestimated.

The risk calculations for the three sites are discussed in greater detail in section 4.1.4 below.

Confidence in data: Uncertainties in input data to calculations of radionuclide transport have been handled rigorously and as far as possible in a uniform manner for all data. The information on the uncertainties is used to formulate risk calculations, but also to evaluate the importance of different factors for the calculation results. The evaluation serves as a basis for an assessment of areas where further research could yield improved knowledge of value for the safety assessment:

- radionuclide turnover in the biosphere,
- fuel dissolution,
- hydraulic description on deposition hole scale.

Confidence in models: The quantification of radionuclide transport provides the risk measure that is directly compared with the Swedish Radiation Protection Institute's acceptance criteria. An evaluation of confidence in the models for radionuclide transport is therefore important.

The requirement on confidence in a model must be formulated in relation to the purpose of its use. For a safety assessment, it is above all important to demonstrate confidence that the models do not underestimate the consequences.

There are several different model concepts for groundwater flow, and three different concepts are compared in SR 97. The conclusion of the comparison is that the natural variability of input data to the models influences the result more than the choice of model.

Confidence in the models for radionuclide transport in canister, buffer and geosphere is judged to be adequate. Many fundamental processes such as diffusion and advection can be given reliable mathematical treatments in the models. The consequences of other processes, such as corrosion of cladding tubes, are simplified by means of pessimistic assumptions. Still others, such as surface diffusion, are handled in a simplified manner by means of pessimistic choices of data. The biosphere model contains rough pessimistic simplifications, especially for the peat ecosystem, which has the greatest consequences.

Climate scenario

The consequence of future climate change is analyzed in the climate scenario. A colder climate can be expected with high probability, but when the changes will occur and how great they will be are difficult to predict. The situation is handled in SR 97 by projecting a future climatic evolution whose main features are governed by astronomical events, but where the quantitative impact on the repository system during different periods is surrounded by greater uncertainties. The sketched evolution embraces a wide span of different climatic conditions. Even though the details of the changes will never be able to be foreseen, confidence is good that the analyzed conditions together cover possible climate changes in a hundred-thousand-year perspective.

The analysis is greatly simplified by the fact that large climate-related changes on the surface at the three repository sites only lead to limited changes at repository depth. This also justifies the format of the analysis of the consequences of climate change, namely as a comparison with repository conditions in the base scenario where the present-day climate is assumed to persist.

The overall conclusion of the climate scenario is that the climatic evolution does not lead to failure of intact canisters. Furthermore, the aggregate effect of the changes in the climate scenario on radionuclide transport processes is such that the consequences in the form of doses are expected to be less than in the canister defect scenario.

Isolation: The conclusion that the canister remains intact is essentially reached by setting bounds on the changes in temperature and rock stresses as well as in the composition, pressure and flow of the groundwater caused by the climate changes at repository depth. The groundwater pressure can be bounded by estimating the maximum ice thickness above the repository. As far as groundwater composition is concerned, there is reason to expect changes in salinity, while oxygen-containing water is not expected to infiltrate down to repository depth other than possibly during very limited periods. Changes in rock stresses can also be bounded based on the weight of the ice cover. The rock stresses do not cause deformations that damage the canisters.

Confidence that future changes in the repository system with regard to isolation will lie within the estimated bounds is good.

Retardation: Climate change also causes changes in transport conditions for radionuclides in buffer and above all geosphere. The changes in transport data almost always lie within the frame of the data used in the canister defect scenario.

Biosphere: The most important change lies in a sharp reduction of the consequences in the biosphere, since the repository sites are covered by ice or water during most of the period with a colder climate. The consequences of initially damaged canisters are thereby **less** than in the canister defect scenario.

Compared with the extreme consequences a northern European glaciation will have on living conditions for human beings, the effects of a deep repository on man and nature appear negligible.

Earthquake/tectonics scenario

A new method for analyzing the consequences of earthquakes with site-specific data on both the geosphere's fracture system and the frequencies and magnitudes of earthquakes is introduced in SR 97. The method represents the first step in the development of a future procedure for quantitative analysis of earthquake scenarios and contains several more highly pessimistic simplifications. Nevertheless, calculations with even this simplified model show that the probability of canister damages during the next 100,000 years is of the same order of magnitude as that assumed for initial canister damages, i.e. a fraction of a percent of the canisters is affected.

The pessimistic assumptions that are made in the risk analysis are judged with good margin to be able to compensate for the uncertainties that are associated with the prediction of future earthquakes. The risk of canister damages caused by earthquakes is therefore negligible.

Intrusion scenarios

The list of human actions that could influence the conditions on a repository site can be made long. Since the future development of human society is basically unpredictable, it can never be made complete. In view of each generation's right to choose how to act under different conditions and their obligation to assume full responsibility for their own actions, a practice has been established that the safety assessment should only deal with inadvertent future actions that can disturb the repository.

Possibilities for quantifying the risks that human beings will intrude into the repository in the future are greatly limited. Nor is it fully clarified how such risks should be taken into account in the total assessment of the acceptance for a deep repository.

In order to reduce the probability that human actions will inadvertently affect the safety of the repository, the site selection process takes account of the fact that different sites may have different potentials for alternative uses, e.g. for mining of ores or unusual minerals. Moreover, the depth of the repository is a way to avoid both natural disturbances on the surface and the effects of human activities.

The probability that inadvertent actions will disturb the repository can also be influenced by the length of the period of institutional controls at the repository site and by how knowledge of the repository can be preserved for the future. However, all analyses of such factors spanning more than a few hundred years run into the difficulty of predicting how society will develop. A specific question concerns the risk of the collapse of society with resultant loss of knowledge and technical competencies.

The question of how possible effects of human actions are to be evaluated, and what responsibility we bear today for guarding against a collapse of society, is strongly influenced by ethical values. A fundamental principle is that safety is best ensured by keeping the radioactive waste contained in isolated and demarcated repositories.

How possible evolutions of society and future human actions that somehow affect the repository can be categorized and expressed as different scenarios is discussed in SR 97.

In an illustrative example, a situation is analyzed where a canister in the repository is inadvertently penetrated by rock drillers. Dose and risk are calculated for the drilling personnel and for a family that settles on the site at a later point in time. The drilling personnel may be exposed to the highest doses, but the risk to both drilling personnel and family are judged to be far below the acceptance criterion, 10^{-6}/year, since the probability of the analyzed events is estimated to be very small.

4.1.3 Weighing-together of scenario analyses

Total risk analysis

The total risk from a deep repository is a summation of the risks associated with all the different future evolutions a repository might undergo.

In a strictly executed risk calculation, the probability of each conceivable evolutionary pathway is estimated and multiplied by the calculated consequence. The sum of all such partial risks is the total risk, which is compared with the Swedish Radiation Protection Institute's acceptance criterion.

Such strict probability estimates cannot be done for a complex system such as a deep repository, whose evolution must be analyzed hundreds of thousands of years into the future. The customary method, which is the one applied in SR 97, is instead to collect a number of possible evolutionary pathways with common main features in a set of scenarios. The probability of each scenario can then be estimated or bounded upward.

With the scenario definitions that apply in SR 97, it would be reasonable to add the consequences of the canister defect and earthquake scenarios to those of the base scenario without weight factors. The canister defect and earthquake scenarios contain in themselves probability factors in the form of frequency of initial and earthquake-induced canister damages, respectively. Both of these factors have been excluded from the base scenario by definition.

In the climate scenario, the effects of initial canister damage are discussed, and to this scenario should also be added the probability of earthquake-induced damage.

The situation is complicated by the fact that it is not possible with present-day knowledge to determine what share of the earthquake statistics in the earthquake scenario is caused by residual effects of previous climate changes and what share has other causes. For this reason, all consequences of the earthquake scenario are added to both the base and climate scenarios.

The result is two "superscenarios" where different effects are added. One describes the course of events when today's climatic conditions persist, and includes both initial canister defects and earthquakes. The other describes what happens when the climate changes, with initial canister defects and earthquakes. The latter must be regarded as a much more probable alternative than the former, since climate change is highly probable.

In SR 97, consequences have only been calculated in the canister defect and climate scenarios. No radiological consequences arise in the base scenario. Nor is the earthquake scenario, where the analysis is preliminary, expected with refined analyses and/or suitable repository layout to have any consequences. The aggregate risk associated with the repository is thereby a weighing-together of the risks in the canister defect and climate scenarios. The risk in the canister defect scenario is calculated for most epochs to be much greater than the risk in the climate scenario, where the biosphere conditions often provide high dilution.

The aggregate risk associated with the repository is therefore estimated pessimistically in SR 97 to be equal to the calculated risk in the canister defect scenario.

According to the Swedish Radiation Protection Institute, consequences associated with intrusion scenarios can be assessed separately from other scenarios, and the intrusion scenarios have therefore not been included in the above line of reasoning.

General assessment of the safety of the KBS-3 method

An in-depth analysis and integrated accounting of the long-term safety of the KBS-3 method for deep geological disposal of spent nuclear fuel has been carried out in SR 97. The results confirm the previous picture that a well-designed repository located in rock with properties that do not differ essentially from normal Swedish rock has good prospects of meeting the regulatory authorities' safety requirements with ample margin.

The KBS-3 system has a flexibility as regards repository depth and layout which allows adaptation to site-specific conditions and to the information on rock conditions which is continuously collected during site investigations and repository excavation.

SKB believes that the repository design that is analyzed in SR 97 has achieved sufficient maturity, that our general understanding of the repository's long-term performance is sufficiently good, and that its potential for high safety has sufficient margins to constitute a satisfactory basis for carrying out site investigations.

4.1.4 How do different conditions in Swedish bedrock affect the feasibility of building a safe repository?

In SR 97, three sites are analyzed to shed light on different conditions in Swedish granitic bedrock as regards geology, groundwater flux, water chemistry, nearness to coast, northerly or southerly siting, surrounding biosphere, etc.

Much information can be obtained from the results of the various scenario analyses to illustrate the importance of varying conditions.

Long-term safety, isolation

Base scenario: The safety margins in the base scenario as far as isolation is concerned are very great for all sites. This applies to both mechanical and chemical stresses on the canister. Groundwater composition varies slightly between the sites, but the differences are unimportant as far as e.g. copper corrosion is concerned. Site-specific rockmechanical analyses have not been carried out; instead, the consequences of the mechanical evolution on the stability of the geosphere around the repository have been bounded by general and very pessimistic approximations.

Climate scenario: The uncertainties regarding the detailed climatic evolutions of the sites are great, but more definite statements can be made about differences between the sites. An important factor for isolation is above all the thickness of an ice cover during a glaciation. The thickest ice cover is expected at Ceberg, where the ice also remains the longest. This means that the mechanical stresses on the canister, as a result of both increased groundwater pressure and mechanical load on the bedrock, are greatest at Ceberg and smallest at Aberg. Changes in the groundwater composition in the climate scenario are not deemed to result in any site-specific differences of importance for isolation.

Earthquakes: As far as the probabilities of earthquake-induced canister damages are concerned, calculations give values that are small and equivalent at the three sites. The calculations in SR 97 have been carried out with pessimistic assumptions that greatly influence the result. It is therefore difficult to draw any conclusions about the relatively suitability of the sites as regards earthquakes. Furthermore, the earthquake risk can be reduced by utilizing a larger part of the rock for the repository. Any difference in terms of earthquake risks would thereby have economic rather than safety-related consequences. Differences between the sites stem from differences in the local fracture structure in combination with the chosen repository placement, as well as from differences in regional earthquake statistics. The earthquake statistics represent one of the greatest uncertainties in the analysis of earthquakes.

Summary, isolation: Conditions at all three sites offer very great safety margins for the repository's isolating capacity in the base scenario.

The evolution in the climate scenario as well is judged to lead to a retention of isolation at all analyzed sites. The safety margin to mechanical canister failure is deemed to be smallest at Ceberg and greatest at Aberg.

A preliminary assessment is that a safe repository in terms of earthquakes can be built at the three sites with optimized site-specific repository layouts. It is not possible today to make any comparisons between the sites in this respect.

Long-term safety, release consequences

Canister defect scenario: The calculations in the canister defect scenario (present-day climatic conditions) show that the Swedish Radiation Protection Institute's acceptance criterion for a deep repository is satisfied with ample margin on all sites. The margin is smallest for Aberg and greatest for Ceberg. The difference pertains above all to the time after 100,000 years. The risk calculations have many pessimistic assumptions to ensure the risk is not underestimated.

If the release takes place to a well, the risk at Aberg is less than one hundredth of the acceptance criterion if the calculation is performed for a timespan of one million years. The risk at Beberg is approximately one-fifth, and at Ceberg one-tenth, of the risk at Aberg.

In the case of release to a peat ecosystem, the sites differ in roughly the same way for times up to around 100,000 years. For a million years, the risk at Aberg then increases by approximately a factor of 40, at Beberg by a factor of 10 and at Ceberg only marginally. The natural radionuclide Ra-226 dominates the consequences for the peat ecosystem for times over 100,000 years.

The calculated retention capacity is equivalent at the three sites for long-lived, non-sorbing nuclides. Retention for sorbing nuclides is strongest at Ceberg and weakest at Aberg.

The differences are primarily due to different hydraulic permeabilities. The rock mass in Aberg and Beberg is, for example, approximately 100 times more permeable than that in Ceberg, with reservation for the fact that the investigation methodology differs between the sites.

Climate scenario: The big difference between the canister defect scenario and the climate scenario as regards the consequences of radionuclide releases lies in the altered and more favourable biosphere conditions. All repository sites are expected to be covered by ice or sea for much of the coming hundred thousand years. Biosphere conditions are deemed to be most favourable at Aberg, which is expected to be sea-covered during most of the period. Ceberg is most positively affected by the climate changes, but here as well the consequences of a radionuclide release will be much less than in the canister defect scenario.

Earthquakes: The evaluation of the analysis of the earthquake scenario indicates that the probability of earthquake-induced canister damages is very small. The release consequences for this scenario are therefore not analyzed.

Summary, release consequences: From this aspect as well, it is deemed possible to build a repository with large safety margins at all sites. The poorer retention at Aberg is compensated for by the more favourable biosphere evolution on the site. It is not meaningful to rank the sites with regard to the consequences of radionuclide release.

Thermal conditions

Thermal conditions differ slightly between the sites as regards temperature at repository depth and thermal conductivity of the rock. The site-specific thermal conditions, together with the fracture structure, determine how large a portion of the host rock must be exploited to accommodate a given quantity of fuel. The governing criterion is that the temperature on the surface of the copper canister may not exceed 100ºC.

The thermal conditions at Aberg, in combination with a relatively limited study site, have led to the proposal of a two-level repository layout, while a single-level repository is proposed in Beberg and Ceberg.

The differences in thermal conditions are primarily of economic importance, since the thermal criterion can always be met if a sufficiently large rock volume is used.

Conclusions

In summary, it is judged that a safe deep repository for spent nuclear fuel in accordance with the KBS-3 method can be built on a site where the conditions resemble those at either Aberg, Beberg or Ceberg.

The safety margins are calculated to be large on all sites. SR 97 has not revealed any differences in long-term safety between the three sites that have any crucial bearing on a weighing-together of all the factors that influence the siting of a deep repository. Such factors include e.g. technology, economics aspects, land use, environmental impact and societal consequences.

4.2 Methodology for safety assessment

According to the methodology description in section 1.8, the assessment can be said to consist of five tasks:

- system description,
- description of initial state,
- choice of scenarios,
- analysis of scenarios,
- evaluation.

Several new approaches have been tried in the methodology in SR 97. The methodology for the four first tasks, as well as handling of uncertainties, is discussed below.

4.2.1 System description

The system description provides a structure for describing the state of the repository system in time and space and the processes that can change this state over time. SR 97 introduces the THMC structure for the system description.

The format for the description provides a more accurate picture of the processes that govern the evolution of the system than previous descriptions using interaction matrices.

Another important advantage is that the same format can be used in the subsequent scenario analyses where the evolution of the system is described in terms of partially coupled thermal, hydraulic, mechanical and chemical evolutions. With previous descriptions it has been difficult to show how the information in the system description has been transferred to the scenario analyses.

The methodology in SR 97 is process-oriented, which is underscored by the fact that the Process Report constitutes a cornerstone in the system description and thereby also in the basis for the subsequent analyses. In the Process Report, all processes are described according to the same format. Experience from the first version of the Process Report used in SR 97 is good. The approach has necessitated a clearer presentation of the knowledge base and permitted a quantification of different processes. The Process Report will need to be revised before each future assessment.

A number of variables that indicate the state of the repository system with time are also established in the system description. This provides a structure for describing the initial state of the system, which is the point of departure of the analysis. Uncertainties in the initial state can thereby be studied systematically and contribute to the basis for scenario selection.

The classification of variables should also be able to be used to provide a stricter description of the evolution of the repository system in future safety assessments.

The work of identifying internal processes and external conditions of importance to repository evolution also belongs to the system description. Conclusions pertaining to this are found in section 4.1.1 above.

4.2.2 Choice of scenarios

The choice of scenarios in SR 97 is an expert assessment based on previous experience, the content of available databases, choice of scenarios in the safety assessments of sister organizations, etc. The choice needs to be revised for future assessments, but this is not expected to lead to any essential changes in the set of scenarios. A clearer coupling between system description, choice of scenarios and information databases should also be made. The choice of scenarios will necessarily involve some expert judgements in future safety assessments as well.

The point of reference in SR 97 is a base scenario where present-day conditions in the surroundings are postulated to persist, despite the fact that climate change is most likely to be expected, at least in a ten-thousand-year perspective. The reason for this choice of base scenario is above all to make a clear distinction between changes caused by internal processes in the repository and those caused by interaction with a changing environment. Furthermore, the Swedish Radiation Protection Institute's regulations expressly require that a case where today's conditions persist be analyzed. The choice of base scenario in SR 97 will also be commented on under the next heading.

The basis for the assessment of probabilities that different scenarios will be realized is weak. Rough, pessimistic estimates similar to those in section 4.1.3 are deemed to be necessary in future assessments as well.

4.2.3 Analysis of chosen scenarios

The frames for the scenario assessments in SR 97 are given by the initial state and the conditions in the surroundings that are defined in each scenario. Based on these premises, the evolution of the repository as a consequence of the internal processes and interaction with the surrounding environment is analyzed. The THMC format has made it possible to show how the complex system of processes can be broken down into a thermal, a hydraulic, a mechanical and a chemical evolution with some essential couplings between them. The format of the system description also makes it possible to show how processes are systematically handled in the analyses.

The point of departure for the scenario analyses is a base scenario. The course of events in other scenarios is then compared with that in the base scenario. This method has been valuable, particularly in the analysis of the canister defect and climate scenarios. Effects and uncertainties caused by internal processes can be distinguished from those caused by external changes. This method is also natural, since the repository system is designed to be robust: Repository performance should not be seriously affected by the changes that can be expected in the surroundings.

The scenario analyses are largely done by means of model calculations. SKB has access to a large set of modelling tools for, among other things, thermal calculations, calculations of groundwater flow, chemical evolution in groundwater and buffer, and radionuclide migration in the near field, geosphere and biosphere. The models are refined continuously as needed. Probabilistic calculations have been utilized in several ways for the analyses in SR 97.

4.2.4 Handling of uncertainties

How qualitative and quantitative uncertainties are handled is an important question that concerns all aspects of a safety assessment. Table 1-1 in section 1.8.2 shows how uncertainties are handled in SR 97.

Completeness

The question of completeness in process identification and choice of scenarios is commented on in section 4.1.1.

Conceptual uncertainty; process understanding and model uncertainty

What is often termed conceptual uncertainty, with varying implications in different contexts, has in SR 97 been divided into the concepts "fundamental process understanding" and "model uncertainty". The former refers to the scientific understanding of a process, the latter to uncertainties that enter when a process is analyzed with a mathematical model in a safety assessment.

Uncertainties surrounding the fundamental understanding of different processes are discussed in the Process Report. The standard format for the process descriptions has been valuable for the accounting of uncertainties, and the descriptions can be further refined in coming versions of the Process Report.

Model uncertainties are discussed where modellings occur in the scenario analyses. This can also generally be carried out more systematically for future assessments. The handling of uncertainties is more sophisticated for the important models used for quantification of radionuclide transport. Three different conceptual models for groundwater flow are used with the important result that the differences in the predictions are small compared with, above all, the natural variability in the bedrock. Confidence in the models for groundwater flow and for radionuclide transport in the near field and geosphere is discussed in greater detail. Confidence that the models do not underestimate the consequences is deemed to be good.

Input data

Uncertainties in input data are discussed in connection with modellings. Here as well, the methodology is most refined for data for radionuclide transport calculations. All these data are compiled in a separate Data Report. The method of choosing reasonable and pessimistic values for all data and then using them in both probabilistic and non-probabilistic analyses has worked well. Among other things, it has been possible to study the influence of different parameters on the repository's retarding function systematically. The method for probabilistic calculations shows a practicable way for carrying out probabilistic analyses where statistical data are lacking. The method is new and needs to be evaluated for use in future assessments.

Summary

The methods that have been tested for the handling of uncertainties in SR 97 have proved to be practical in actual use. The application of the method can be refined in future assessments. The new methods that have been introduced for accounting and handling of data uncertainties for the quantification of radionuclide transport are judged to be able to serve as a basis for future safety reports as well.

4.2.5 Assessment of available methodology

SKB finds, with the support of the above account, that the methodology used in SR 97 constitutes a sufficient basis for future safety assessments based on data from site investigations.

4.3 Basis for site selection and site investigations

SR 97 constitutes an important basis for the ongoing work aimed at formulating and quantifying requirements and preferences regarding the host-rock from the perspective of long-term repository safety. Experience from SR 97 is also used in the work of formulating an integrated programme for investigation and evaluation of sites.

4.3.1 What requirements does the deep repository make on the host-rock?

SKB has pursued the work of formulating and quantifying requirements and preferences regarding the rock in parallel with SR 97. The results will be reported in 2000, and a progress report was made in conjunction with the publication of RD&D-Programme 98. The project identifies so-called geoscientific suitability indicators based on an analysis of what requirements and preferences can be formulated regarding conditions in the rock and properties of the rock. The requirements can be made both with a view towards long-term safety and so that it will be technically possible to build the repository. The judgements of what is essential from the viewpoint of long-term safety are based on previous knowledge and experience and on the analyses that have been carried out within the framework of SR 97.

A first step in the work involved determining what geoscientific information is used in safety assessment and construction analysis. Such an inventory was reported in 1996 and has now been checked against the more comprehensive analysis that was done in SR 97. In a safety assessment, it is primarily the information on the initial state that must be taken from site-specific investigations.

Since the integrated performance of the deep repository in different time periods is dependent on a large number of interacting processes, it is difficult to specify more detailed requirements on individual initial conditions. On the other hand, there is a need for a structure for the siting work and guidance as to what is essential to measure in a site investigation. For this reason, all the site-specific properties of a host rock have been reviewed and the question has been asked for each one as to whether there are value ranges that would seriously degrade any of the deep repository's isolating or retarding functions.

If it becomes apparent that there are value ranges where the deep repository's isolation may be seriously threatened, the requirement is made that these parameter values may *not* occur. This knowledge is obtained above all from the analysis of the base scenario. Conditions that could threaten e.g. canister integrity, for example the presence of dissolved oxygen, are not accepted and thereby become the subject of requirement formulations. The requirements are not absolute in the sense that the deep repository would definitely be unsafe if the requirements were not met, but are nevertheless made as a precaution. The requirements can only be reconsidered in the light of new knowledge or after a change in the repository's design.

Even if it is not possible to find grounds for requirements, it is often possible to formulate firm preferences regarding value ranges that contribute to good isolation or good retardation. The preferences stipulate value ranges that are conducive to the desired function, but do not have to specify the limit to unacceptable function. Such a limit is influenced in many cases by other conditions, is relative, is unknown or can be influenced by repository layout.

Formulation of preferences has been based on the value ranges analyzed in SR 97, along with consideration of whether other value ranges might lead to better function. The knowledge is taken from the detailed analyses of especially the base scenario and the canister scenario. In a similar manner, preferences regarding the rock are formulated from the perspective of constructability/technology.

The requirements and preferences have been formulated to provide guidance in the siting work and to permit selection of investigation methods used in site investigations. They do not take the place of integrated and comprehensive safety assessments.

4.3.2 Programme for site investigations

The identification of geoscientific suitability indicators is in turn one of the most important points of departure for SKB's programme for investigation and evaluation of sites, which will be presented in 2000. The investigation will provide a basis for judging value ranges for essential geoscientific factors on candidate sites. Criteria for site evaluation are also used to prioritize the scope of the investigations and permit more rapid feedback to the safety assessment when investigation results become available. The biosphere on the site is also surveyed.

The site investigation programme includes more than the information sought by the safety assessment, however. The investigations are supposed to provide a basis for a general geoscientific understanding, and many investigations do not in themselves furnish direct data for analyses, but are used when data are to be interpreted. These questions are also dealt with and discussed when the site investigation programme is formulated.

4.4 Basis for functional requirements

According to SKI's requirements on SR 97, the safety assessment should also serve as a basis for "deriving preliminary functional requirements on the canister and the other barriers."

The safety assessment can do this in a general sense. In analyzing the different scenarios, it describes what external stresses the repository system may be exposed to. The repository must be designed to function under these stresses, which is a very general formulation of functional requirements. The requirement applies to the system as a whole. It is only possible in certain cases to directly derive requirements for individual barriers from conditions in the surroundings. An example where this is possible:

The canister has to withstand the hydrostatic pressures and swelling pressure that occur in the base scenario, as well as the maximum hydrostatic pressures that can result from a glaciation. Both of these pressures are quantified in SR 97. The assessment thereby provides a basis for formulating functional requirements on the canister, which must retain its isolating function at these pressures.

It is more difficult to formulate functional requirements for tectonic changes. The effect of an earthquake can be controlled via the repository layout, and even if the layout has been established the effects of an earthquake are dependent on the design of the canister-buffer combination. The mechanical effects on the canister are determined by e.g. buffer thickness, and it is the combination of the two barriers that determine their performance.

Functional requirements for the buffer can be derived from the groundwater composition in the different scenarios: The buffer should retain its intended function when the groundwater has the compositions that may be encountered on the repository site during different time periods in both the base and canister defect scenarios. SR 97 provides a quantitative basis for this in the form of estimates of the different groundwater compositions that may be encountered in Swedish bedrock.

Functional requirements for the canister are not as simple to derive from the groundwater composition: Buffer and canister together determine how well the canister's isolating function withstands the chemical stresses. Functional requirements for the canister cannot be derived directly from the results of the assessment; however, the function of the canister-buffer system must be such that canister isolation is not jeopardized in a long time perspective.

The safety assessment quantifies the conditions in the surroundings and then talks about whether the chosen system design works under these conditions. However, the analysis methodology is not in general aimed at analyzing which detailed properties the barriers must possess in order to "withstand" the stresses. This inverse problem does not even have a straightforward solution.

SR 97 thus furnishes quantitative material in the form of groundwater compositions, flows and pressures, rock movements and other data that can be used as a basis for modifying the repository design if necessary, but it is only in a few cases that functional requirements can be derived directly from the analysis results.

Review of repository design

The design of the deep repository is based on the primary requirement that the repository should isolate the waste from man and the environment. If this isolation should for some reason be breached, the repository should have a secondary retarding function. Based on these fundamental functional requirements, more detailed functional requirements are established for the canister and the other barriers. Aside from the requirements on long-term safety, there are requirements aimed at facilitating fabrication, construction and operation.

The detailed repository design will be refined gradually in a long-term process, and in the meantime a number of safety assessments will be carried out. The results from each assessment will be fed back to the repository design work and may lead to modifications of functional requirements, design premises and the design/layout of different parts. Changes in design details are also initiated by experience from the development of technology for the execution of the different parts of the repository. The modified design will then serve as the point of departure for the next safety assessment.

The result of the work with SR 97 will be used as a basis for a review of the functional requirements and design premises that determine the design of the canister and the other barriers in the deep repository.

4.5 Prioritization of research

The experience gained in SR 97 also serves as a basis for refining the methodology for future safety assessments, as well as for setting priorities in the programmes of supportive research, development and technology demonstration.

Methodology for safety assessment

The programme for refinement of safety assessment methodology in RD&D-98 will be evaluated in the light of the experience from SR 97 and its review. Needs for future safety assessments that have been identified in SR 97 include:

- a revised Process Report,
- a study of the possibilities of a more systematic choice of scenarios, and
- an evaluation of probabilistic calculation methods.

Supportive research

The results of SR 97 can be used in the programme for supportive research to set priorities for R&D work on deep disposal of spent nuclear fuel. Besides annual programme reviews, a more far-reaching audit will be carried out in connection with RD&D-Programme 2001. The results of SR 97 indicate several areas that may need to be prioritized, for example:

- biosphere modelling,
- earthquake modelling,
- long-term effects of creep movements in the rock,
- the mechanical effects of tectonic rock movements on the canister, e.g. creep effects in the copper shell,
- general function of the backfill,
- erosion of buffer and backfill under different climatic conditions,
- the early hydromechanical evolution of the canister-buffer gap,
- models for hydrology and radionuclide transport on a detailed scale around deposition holes to permit optimal choices of deposition holes,
- fuel dissolution.

The needs have been identified either directly in the Process Report or when the integrated evolution resulting from several coupled processes has been studied in the safety assessment. New findings would often lead to less pessimistic treatments in the safety assessment.

Quality control

Quality control is essential in order to ensure the reliability of the assessments. Quality requirements can be made on data, models and evaluations. Traceability and the ability to reproduce results are other important aspects. A quality-controlled system for data collection and data preservation has been tested at the Äspö HRL. A version management system has been utilized for models of groundwater flow and radionuclide transport in SR 97. Premises, input data and results are archived for these calculations either digitally, with the customary security routines, or as hard copies in accordance with SKB's archive rules.

The procedures for quality control of the safety assessment need to be refined. SKB intends to have the company certified with a complete QA system in accordance with ISO 9001 when the site investigations begin. The safety assessment is covered by the certification.

4.6 Closing words

The next stage in the siting of a deep repository entails investigation of the bedrock at a number of candidate sites for a repository. The main purpose of SR 97 is, in preparation for this next stage, "to demonstrate that the KBS-3 method has good prospects of being able to meet the safety and radiation protection requirements which SKI and SSI have specified in recent years."

The radiation levels which Swedish authorities accept for individuals in the vicinity of a deep repository lie around one percent of the natural background radiation. The results of the analyses in SR 97 show maximum levels that are less than one-tenth of the official limits. The maximum levels will occur tens of thousands of years in the future and during the relatively short time intervals when candidate repository sites in Sweden are not expected to be covered by ice sheets or sea.

The results should be regarded in the light of the cautious attitude that permeates the execution of the safety assessment. Wherever knowledge in a field is not complete, a poorer outcome than is reasonable to expect is pessimistically assumed.

SR 97 shows that the prospects for building a safety deep repository for spent nuclear fuel in Swedish granitic bedrock are very good. The assessment is comprehensive and detailed by international standards when considered in the light of the next stage SKB is facing.

It is SKB's judgement that the scope of the safety assessment and confidence in its results satisfy the requirements that should be made in preparation for such a stage.