Technical Report

TR-11-01

Long-term safety for the final repository for spent nuclear fuel at Forsmark

Main report of the SR-Site project

Volume I

Svensk Kärnbränslehantering AB

March 2011

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Keywords: Safety assessment, Long-term safety, Final repository, Spent nuclear fuel, Forsmark.

Update notice

The original report, dated March 2011, was found to contain both factual and editorial errors which have been corrected in this updated version. The corrected factual errors are presented below.

Updated 2015-05

No updates in Volume I.

Updated 2012-12

Location	Original text	Corrected text
Page 58, paragraph 1, line 4	...Canister production report /SKB 2010a/, see...	...Canister production report, see...
Page 246, Table 7-8, column 5	SKB 2006	SKB 2006c

Updated 2011-12

Location	Original text	Corrected text
Page 179, Figure 5-11	Width of pellet filled gap 60 mm	Figure 5-11 updated Width of pellet filled gap 50 mm

Updated 2011-10

Location	Original text	Corrected text
Page 38, paragraph 4 from bottom	...value below background radiation.	...value below 1 mSv/hour.
Page 67, paragraph 4	...in the scenario selection in a....	...in the scenario analyses in a....
Page 67, paragraph 5	This is described briefly in Section 6.2.1 and in more detail when applied in the scenario selection, Chapter 11 and the analysis of FHA scenarios, Section 14.2.	This is described in the analysis of FHA scenarios, Section 14.2.
Page 97, paragraph 3, line 3	...in Chapter 13.	...in Chapter 11.
Page 111, text to figure 4-8, last line	...of the target area,	...of the candidate area,
Page 168, Table 5-8	Wrong data in table	Table updated with correct data
Page 186, Table 5-15, column 1	(mm)	(m)
Page 245, Table 7-7, column 3	FARF31	FARF31, MARFA
Page 246, Table 7-8, column 3	DarcyTools	PhreeqC
Page 246, Table 7-8, column 3	DarcyTools	ConnectFlow
Page 246, Table 7-8, column 3	FARF31	FARF31, MARFA
Page 259, paragraph 2, last sentence	...material properties of these components, (see Section 5.5.3 and, for details, /Karnland et al. 2006/) and since, in particular for the backfill, alternative materials are to be evaluated in the assessment, no specific criterion is given here.	...material properties of these components (see Section 5.5.3 and, for details, /Karnland et al. 2006/).
Page 269, Figure 8-4	Arrow from box “Rock stresses” to “Shear at deposition hole?”	Figure 8-4 updated Arrow from box “Fracture structure in host rock” to “Shear at deposition hole?”.

Preface

This document is the main report of the SR-Site project, an assessment of long-term safety for a KBS-3 repository at Forsmark. The report supports SKB's licence application for a final repository for spent nuclear fuel at Forsmark.

The undersigned has been the main editor of the report and has been responsible for the methodology development in consultation with mainly Johan Andersson, JA Streamflow AB and Kristina Skagius, Kemakta Konsult AB. Kristina Skagius has compiled the description of the Forsmark site in Chapter 4 and is responsible for the development of the SR-Site FEP database and for the handling of issues relating to future human actions. Johan Andersson has acted as a co-ordinator of the repository engineering and the safety assessment projects and co-ordinated the descriptions of the initial state of the repository in Chapter 5, of the reference evolution in Chapter 10 and of the feedback regarding design and site related issues in Chapter 15.

The following persons, SKB employees unless otherwise noted, have had the main responsibilities for specific subject areas in the assessment and have provided the corresponding texts in this report: Kastriot Spahiu and Lena Zetterström Evins (fuel); Christina Lilja (canister); Patrik Sellin (buffer, backfill and closure); Jan-Olof Selroos and Sven Follin, SF GeoLogic AB (hydrogeology); Jan-Olof Selroos and Scott Painter, LANL, US (geosphere transport); Raymond Munier and Johan Andersson, JA Streamflow AB (geomechanical issues); Ignasi Puigdomenech and Birgitta Kalinowski (geochemistry); Tobias Lindborg, Ulrik Kautsky and Eva Andersson, Studsvik Nuclear AB (biosphere); Jens-Ove Näslund (climate issues), Lena Zetterström Evins (natural analogues) and Maria Lindgren, Kemakta Konsult AB, Christina Greis and the undersigned (integrated radionuclide transport modelling). Martin Löfgren, Niressa AB has been responsible for the compilation of input data for the assessment in collaboration with Fredrik Vahlund.

The report has been reviewed by SKB's international site investigation expert review group (Sierg), extended with experts on safety assessment methodology: Per-Eric Ahlström, SKB (chair); Lucy Bailey, NDA, UK; Jordi Bruno Amphos21, Spain; John Cosgrove, Imperial college, UK; Tom Doe, Golder Ass. Inc., US; Alan Hooper, Alan Hooper Consulting Ltd, UK; John Hudson, Rock Engineering Consultants, UK; Ivars Neretnieks, Royal Institute of Technology, Sweden; Roland Pusch, Drawrite AB, Sweden; Jürg Schneider, Nagra, Switzerland, Lars Söderberg, SKB, Mike Thorne, Mike Thorne and Associates Ltd, UK and Timo Äikäs, Posiva OY, Finland. It has also been reviewed by Olle Olsson, SKB.

Stockholm, March 2011

Allan Hedin
Project leader SR-Site

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Summary

The central conclusion of the safety assessment SR-Site is that a KBS-3 repository that fulfils long-term safety requirements can be built at the Forsmark site. This conclusion is reached because the favourable properties of the Forsmark site ensure the required long-term durability of the barriers of the KBS-3 repository. In particular, the copper canisters with their cast iron inserts have been demonstrated to provide a sufficient resistance to the mechanical and chemical loads to which they may be subjected in the repository environment.

The conclusion is underpinned by:

- The reliance of the KBS-3 repository on i) a geological environment that exhibits long-term stability with respect to properties of importance for long-term safety, i.e. mechanical stability, low groundwater flow rates at repository depth and the absence of high concentrations of detrimental components in the groundwater, and ii) the choice of naturally occurring materials (copper and bentonite clay) for the engineered barriers that are sufficiently durable in the repository environment to provide the barrier longevity required for safety.
- The understanding, through decades of research at SKB and in international collaboration, of the phenomena that affect long-term safety, resulting in a mature knowledge base for the safety assessment.
- The understanding of the characteristics of the site through several years of surface-based investigations of the conditions at depth and of scientific interpretation of the data emerging from the investigations, resulting in a mature model of the site, adequate for use in the safety assessment.
- The detailed specifications of the engineered parts of the repository and the demonstration of how components fulfilling the specifications are to be produced in a quality assured manner, thereby providing a quality assured initial state for the safety assessment.

The detailed analyses demonstrate that canister failures in a one million year perspective are rare. Even with a number of pessimistic assumptions regarding detrimental phenomena affecting the buffer and the canister, they would be sufficiently rare that their cautiously modelled radiological consequences are well below one percent of the natural background radiation.

S1 Purposes and general prerequisites

The purpose of the safety assessment SR-Site is to investigate whether a safe spent nuclear fuel repository of the KBS-3 type can be built at the Forsmark site in the municipality of Östhammar, Sweden. The Forsmark site has been selected based on findings emerging from several years of surface based investigations of the conditions at depth at the Forsmark site and at the Laxemar site in the municipality of Oskarshamn. The site selection is not justified in the SR-Site assessment, but in other documents supporting SKB's licence application.

The SR-Site report is a main component in SKB's licence application to construct and operate a final repository for spent nuclear fuel at Forsmark in the municipality of Östhammar. Its role in the application is to demonstrate long-term safety for a repository at Forsmark.

Several decades of research and development has led SKB to put forward the KBS-3 method for the final stage of spent nuclear fuel management. In this method, copper canisters with a cast iron insert containing spent nuclear fuel are surrounded by bentonite clay and deposited at approximately 500 m depth in groundwater saturated, granitic rock, see Figure S-1. The purpose of the KBS-3 repository is to isolate the nuclear waste from man and the environment for very long times. Around 12,000 tonnes of spent nuclear fuel is forecasted to arise from the currently approved Swedish nuclear power programme (where the last of the 10 operating reactors is planned to end operation in 2045), corresponding to roughly 6,000 canisters in a KBS-3 repository.

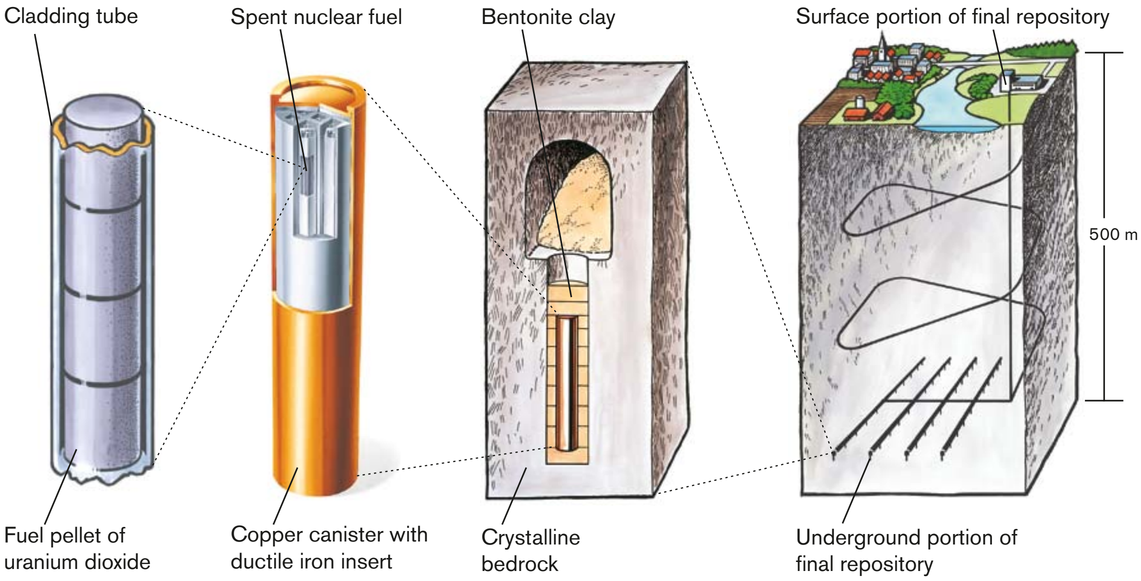


Figure S-1. *The KBS-3 concept for disposal of spent nuclear fuel.*

The main purposes of the safety assessment project SR-Site are:

- To assess the safety, as defined in applicable Swedish regulations, of the proposed KBS-3 repository at Forsmark.
- To provide feedback to design development, to SKB's RD&D Programme, to detailed site investigations and to future safety assessment projects.

An important step leading up to the present report was the preparation of the SR-Can safety assessment report, published in November 2006. The SR-Can report was reviewed by the Swedish safety authorities aided by a group of international experts, and the outcome of the review has been taken into account in the SR-Site assessment.

Regulations

Society's requirements on long-term safety of nuclear waste repositories are ultimately expressed in legal regulations. Two detailed regulations are issued by the Swedish Radiation Safety Authority (SSM) under the Nuclear Activities Act and the Radiation Protection Act, respectively:

- "The Swedish Radiation Safety Authority's regulations concerning safety in final disposal of nuclear waste" (SSMFS 2008:21).
- "The Swedish Radiation Safety Authority's Regulations concerning the Protection of Human Health and the Environment in connection with the Final Management of Spent Nuclear Fuel or Nuclear Waste" (SSMFS 2008:37).

These two documents are reproduced in their entirety in Appendix A to this report. The way in which this SR-Site report addresses the requirements is indicated by references to relevant sections of this report, as inserts in the regulatory texts in the Appendix.

The principal acceptance criterion, expressed in SSMFS 2008:37, concerns the protection of human health and requires that "the annual risk of harmful effects after closure does not exceed 10^{-6} for a representative individual in the group exposed to the greatest risk". "Harmful effects" refers to cancer and hereditary effects. The risk limit corresponds to an effective dose limit of about $1.4 \cdot 10^{-5}$ Sv/yr. This, in turn, corresponds to around one percent of the effective dose due to natural background radiation in Sweden. Furthermore, the regulation SSMFS 2008:21 require descriptions of the evolution of the biosphere, geosphere and repository for selected scenarios; and evaluation of the environmental impact of the repository for selected scenarios, including the main scenario, with respect to defects in engineered barriers and other identified uncertainties.

The timeframe for the assessment – one million years

In the General Guidance to SSM 2008:37, it is indicated that the time scale of a safety assessment for a final repository for spent nuclear fuel should be one million years after closure. A detailed risk analysis is required for the first thousand years after closure. Also, for the period up to approximately one hundred thousand years, the reporting is required to be based on a quantitative risk analysis.

For the period beyond one hundred thousand years, the General Guidance states that a strict quantitative comparison of calculated risk in relation to the criterion for individual risk in the regulations is not meaningful. Rather, it should be demonstrated that releases from both engineered and geological barriers are limited and delayed as far as reasonably possible using calculated risk as one of several indicators.

The hazard of the waste

After approximately 100,000 years, the radiotoxicity of the spent nuclear fuel is comparable with that of the natural uranium ore once used to produce the fuel. Furthermore, the sum of toxicity of all fractions originating from the nuclear fuel cycle (the daughter nuclides separated from the uranium prior to enrichment, the depleted uranium arising in the enrichment process and the spent fuel) is comparable to that of the utilised uranium ore after 100,000 years, see Figure S-2.

It is also noted that the initially very high dose rates from potential exposure to direct, external radiation from the spent fuel decrease substantially within a few thousand years. In the long term, these dose rates will however remain at levels requiring shielding from humans practically indefinitely, since the long-term direct radiation levels is determined by U-238 progeny.

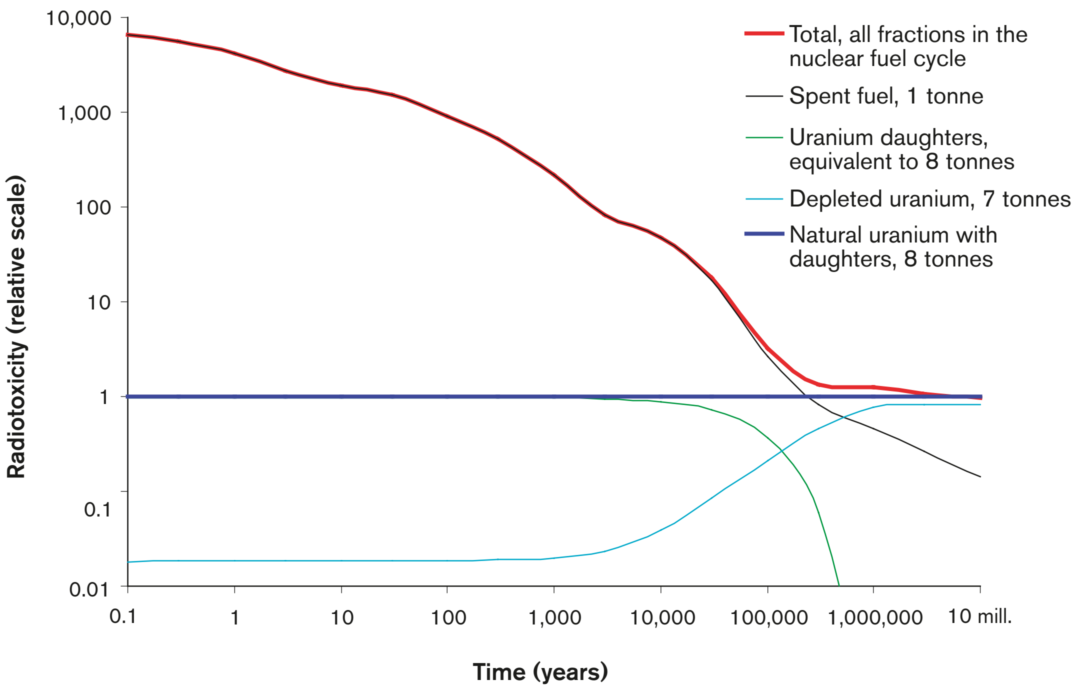


Figure S-2. *Radiotoxicity on ingestion of uranium and daughters in ore (blue line), and of the sum of all fractions that arise when the same quantity of uranium is used in the nuclear fuel cycle (red line). The time refers to the time after reactor operation. The different fractions comprise the spent fuel (38 MWd thermal energy/kg U of type SVEA 64 BWR), the depleted uranium and the uranium daughters that are separated in the uranium mill.*

The stepwise development of the repository programme

The design and safety evaluation of a repository concept for geological disposal like the KBS-3 system is developed in steps, where a safety evaluation in one step provides feedback to the development of the repository design. The developed design is then evaluated in a subsequent safety assessment, which provides refined feedback to the further development of the design, etc. Likewise, the understanding of natural processes of importance to long-term safety is developed in a R&D programme and the emerging findings are evaluated in an iterative interaction with safety assessment projects. Another important aspect of this iterative nature of the development is the external reviewing, by authorities and international experts, of the safety assessments.

SKB has conducted research and development of the KBS-3 system for three decades and both the repository design and the scientific knowledge is mature, as manifested by the facts that no major design changes have occurred in recent years and that the identified set of processes of importance for long-term safety is stable, as is the knowledge about the processes.

SKB has established a technically feasible reference design and layout of the KBS-3 repository and showed that this conforms to the established design premises, see below, but technical development will continue. Detailed designs adapted to an industrialised process designed to fulfil specific requirements on quality, cost and efficiency need still be developed. The layout needs to be adapted to the local conditions found when constructing the repository at depth. These, potentially more optimal solutions, should result in at least the same level of safety as the current reference design being assessed in SR-Site. Since SR-Site is an important basis for a critical decision point in the repository programme, it is essential to demonstrate i) that the essential safety related features of the design are mature and ii) that there is at least one available and adequate option for parts of the system that are more peripheral in terms of contributing to safety.

Another characteristic of the present situation is that the well-established parts of the design are specified in detail; the feedback to design development from the safety assessment preceding SR-Site (the SR-Can assessment) is given in the form of detailed design premises, that have served as input to specifications of the reference design and facilitated the evaluation of the appropriateness of the design with respect to long-term safety.

S2 Achieving safety in practice – the properties of the site and the design and construction of the repository

S2.1 Safety principles

Since work on the Swedish final repository project commenced at the end of the 1970s, SKB has established a number of principles for the design of a final repository. The principles can be said to constitute the safety philosophy behind the KBS-3 concept. They are summarised below.

- By placing the repository at depth in a long-term stable geological environment, the waste is isolated from the human and near-surface environment. This means that the repository is not strongly affected by either societal changes or the direct effects of long-term climate change at the ground surface.
- By locating the repository at a site where the host rock can be assumed to be of no economic interest to future generations, the risk of human intrusion is reduced.
- The spent fuel is surrounded by several engineered and natural safety barriers.
- The primary safety function of the barriers is to contain the fuel within a canister.
- Should containment be breached, the secondary safety function of the barriers is to retard a potential release from the repository.
- Engineered barriers shall be made of naturally occurring materials that are stable in the long term in the repository environment.
- The repository shall be designed and constructed so that temperatures that could have detrimental effects on the long-term properties of the barriers are avoided.

- The repository shall be designed and constructed so that radiation induced processes that could have detrimental effects on the long term behaviour of the engineered barriers or of the rock are avoided.
- The barriers should be passive, i.e. they should function without human intervention and without artificial supply of matter or energy.

Together with many other considerations, like the geological setting in Sweden and the requirement that the repository must be feasible to construct from a technical point of view, these principles have led to the development of the KBS-3 system for spent nuclear fuel.

In practice, safety is achieved through the selection of a site with favourable properties for long-term safety and through the design and construction of a repository that fulfils requirements related to long-term safety. The site conditions today and the design and layout of the KBS-3 repository at Forsmark constitute the initial state of the safety assessment. These are also the aspects that are controlled by the implementer, through the choice of the site and through the design and site adaptation of the repository.

S2.2 The Forsmark site

The Forsmark site is located in the northern part of the county of Uppland within the municipality of Östhammar, about 120 km north of Stockholm. The Forsmark area consists of crystalline bedrock that belongs to the Fennoscandian Shield and formed 1.85 to 1.89 billion years ago. Tectonic lenses, in which the bedrock is less affected by ductile deformation, are enclosed in between ductile high-strain belts. The candidate area is located in the north-westernmost part of one of these tectonic lenses. This lens extends from north-west of the Forsmark nuclear power plant south-eastwards to the area around Öregrund (Figure S-3).

Three major sets of deformation zones with distinctive orientations have been recognized. In addition to vertical and steeply dipping zones, there are also gently south-east- and south-dipping zones. These gently dipping zones are more frequent in the south-eastern part of the candidate volume and have higher hydraulic transmissivity than vertical and steeply dipping deformation zones at the site. The frequency of open and partly open fractures is very low below approximately 300 m depth compared to what is observed in the upper part of the bedrock in the north-western part of the candidate volume, which is the target volume for the repository. In addition, the rock stresses are relatively high compared to typical values of the Swedish bedrock. The upper 100 to 150 m of the bedrock overlying the target volume contains many highly transmissive fractures in the horizontal plane and in good hydraulic contact over long distances, whereas at depth the rock has very low permeability with few transmissive fractures. At repository depth (c. 470 m) the average distance between transmissive fractures is more than 100 m.

Groundwaters in the uppermost 100 to 200 m of the bedrock display a wide range of chemical variability, with chloride concentrations in the range 200 to 5,000 mg/L suggesting influence of both brackish marine water and meteoric waters. At depths between 200 and 800 m, the salinity remains fairly constant (5,000–6,000 mg/L) and the water composition indicates remnants of water from the Littorina Sea that covered Forsmark between 9,500 and 5,000 years ago. At depths between 800 and 1,000 m, the salinity increases to higher values.

Data from site investigation to safety assessment

The site investigation at Forsmark, including processing of the emerging data and site modelling, was carried out between 2002 and 2008. The gathering of information and its transfer from the site investigations at Forsmark to the safety assessment application has involved several steps.

- Field data have been obtained from various investigation activities like airborne and ground geophysics, borehole drilling and borehole testing. After quality control, the data have been entered into the SKB data bases.
- The field data have been interpreted and evaluated into a cross-disciplinary site descriptive model (SDM), being a synthesis of geology, rock mechanics, thermal properties, hydrogeology, hydrogeochemistry, bedrock transport properties and surface system properties, see Figure S-4. The SDM provides a description of the understanding of the site properties within the different

disciplines and it also provides an assessment of the uncertainty in these descriptions. The SDM for the Forsmark site at the completion of the surface-based investigations is reported in a main site-description report and a number of supporting reports.

- The site description and references therein cannot always be used directly in the safety assessment. There is a need to also consider non-site specific information, to add judgements on how to handle the uncertainties identified in the site description and to make final selections of model input data. For this reason, all site data used in SR-Site are assessed in the SR-Site **Data report**, using the SDM as input. The role of the **Data report** is explained in Section S3.7.

As part of the site-descriptive modelling, the uncertainty and confidence in the Forsmark site description were assessed. This assessment comprised exploring confidence in the site characterisation data, key remaining uncertainties in the site description, alternative models and their handling, consistency between disciplines and the main reasons for confidence or lack of confidence in the site descriptive model. The overall outcome of this assessment was that it was found that the site properties of importance for both repository constructability and long-term safety are sufficiently bounded by quantitative uncertainty estimates or alternative models.

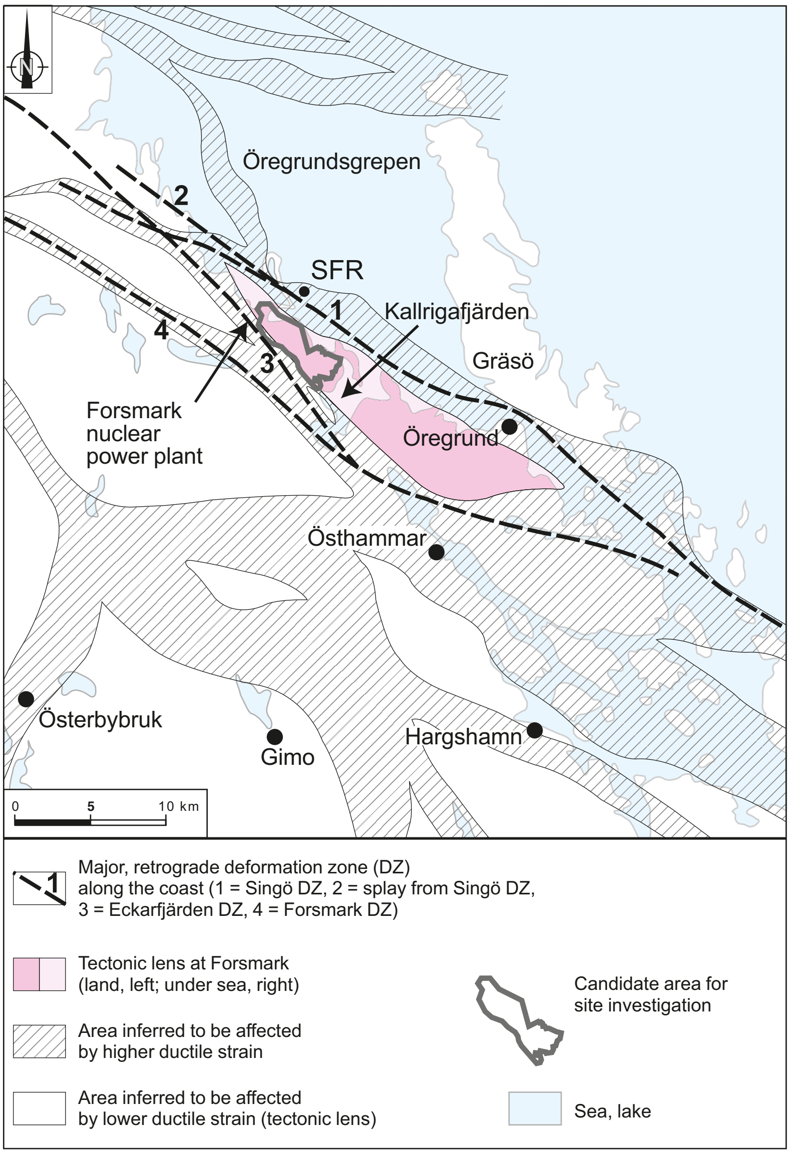


Figure S-3. *Tectonic lens at Forsmark and areas affected by strong ductile deformation in the area close to Forsmark.*

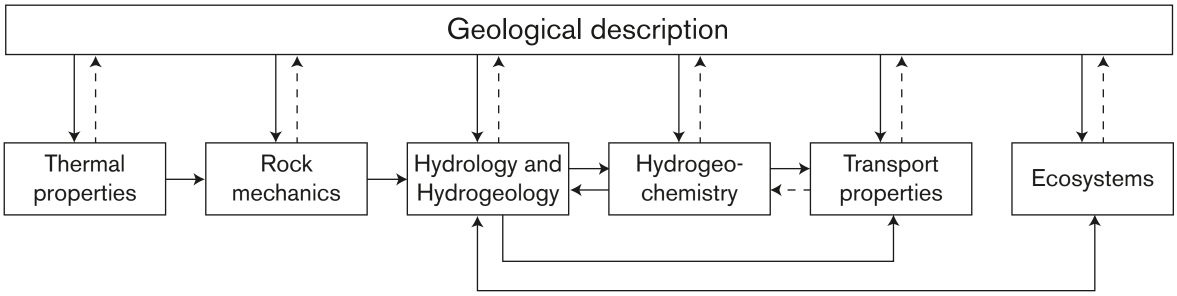


Figure S-4. *The different discipline descriptions in the SDM are interrelated with several feedback loops and with geology providing the essential geometrical framework.*

In summary, the main safety related features of the Forsmark site are:

- A low frequency of water conducting fractures at repository depth.
- Favourable chemical conditions, in particular reducing conditions at repository depth, (which is generally found at depth in granitic rocks in Sweden) and salinity that would ensure stability of the bentonite clay buffer.
- The absence of potential for metallic and industrial mineral deposits within the candidate area at Forsmark.

In addition, the relatively high thermal conductivity at the site facilitates an efficient use of the rock volume and the rock mechanics and other properties of importance for a safe and efficient construction of the repository are also favourable.

S2.3 The site adapted repository reference design

A comprehensive description of the initial state of the repository system is one of the main bases for the safety assessment. The initial state in SR-Site is defined as the state at the time of deposition/installation for the engineered barrier system and the natural, undisturbed state at the time of beginning of excavation of the repository for the geosphere and the biosphere. (Excavation induced impacts on the geosphere and the biosphere are analysed as part of the safety assessment.)

Design premises, reference design and Production reports

The KBS-3 repository concept has been developed since it was first introduced. The current design is based on the design originally presented in the KBS-3 report in 1983. Feedback from assessments of long-term safety is a key input to the refinement of the design. Feedback from the SR-Can assessment was further developed into *design premises* for the SR-Site assessment and the licence application. Design premises typically concern specification on what mechanical loads the barriers must withstand, restrictions on the composition of barrier materials or acceptance criteria for the various underground excavations. Close to 30 different design premises on the canister, the buffer, the deposition holes, the deposition tunnels and backfill and on the main tunnels, transport tunnels, access tunnels, shafts, central area and closure were developed based on the SR-Can assessment and some subsequent analyses. The resulting design premises constitute design constraints, which, if all fulfilled, form a good basis for demonstrating repository safety.

A reference design conforming to the design premises has been developed and is reported in a number of so called **Production reports**. These reports covering the spent fuel, the canister, the buffer, the tunnel backfill, the repository closure and the underground openings contain the information required for the SR-Site assessment of engineered components of the repository system.

Each report gives an account of i) the design premises to be fulfilled, ii) the reference design selected to achieve the requirements, iii) verifying analyses that the reference design does fulfil the design premises, iv) the production and control procedures selected to achieve the reference design, v) verifying analyses that these procedures do achieve the reference design and vi) an account of the achieved initial state. The latter point is the key input to the safety assessment.

The initial state, as given in the **Production reports**, provides quantitative information on key inputs to the safety assessment. These are critically evaluated in the **Data report** where the formal qualification of input data to the safety assessment occurs based on an evaluation of uncertainties affecting the initial state data.

The following is a brief account of key features of the repository design.

Fuel

The major part of the nuclear fuel to be deposited consists of spent fuel from the operation of the twelve Swedish nuclear power plants, which are either of boiling water reactor (BWR) type or pressure water reactor (PWR) type. The fuel types and amounts are derived from the spent fuel stored in Clab (31 December 2007) and a reference scenario for the future operation of the ten remaining power plants. In the reference scenario the operating times are set to 50 years for the four reactors at Ringhals and the three at Forsmark, and 60 years for the three reactors at Oskarshamn. The two reactors in Barsebäck were closed down after approximately 24 years and 28 years of operation, respectively. The majority of the fuel used in the reactors consists of uranium oxide fuel (UOX). From Oskarshamn, there will be minor amounts of mixed oxide fuel (MOX). There are also minor quantities of other oxide fuel types from research and the early part of the nuclear power programme to be deposited in the KBS-3 repository.

Canister

The reference design of the canister consists of a tight, 5 cm thick corrosion barrier of copper and a load-bearing insert of nodular cast iron. The sealed canister has a total length of 4,835 mm and a diameter of 1,050 mm, see Figure S-5.

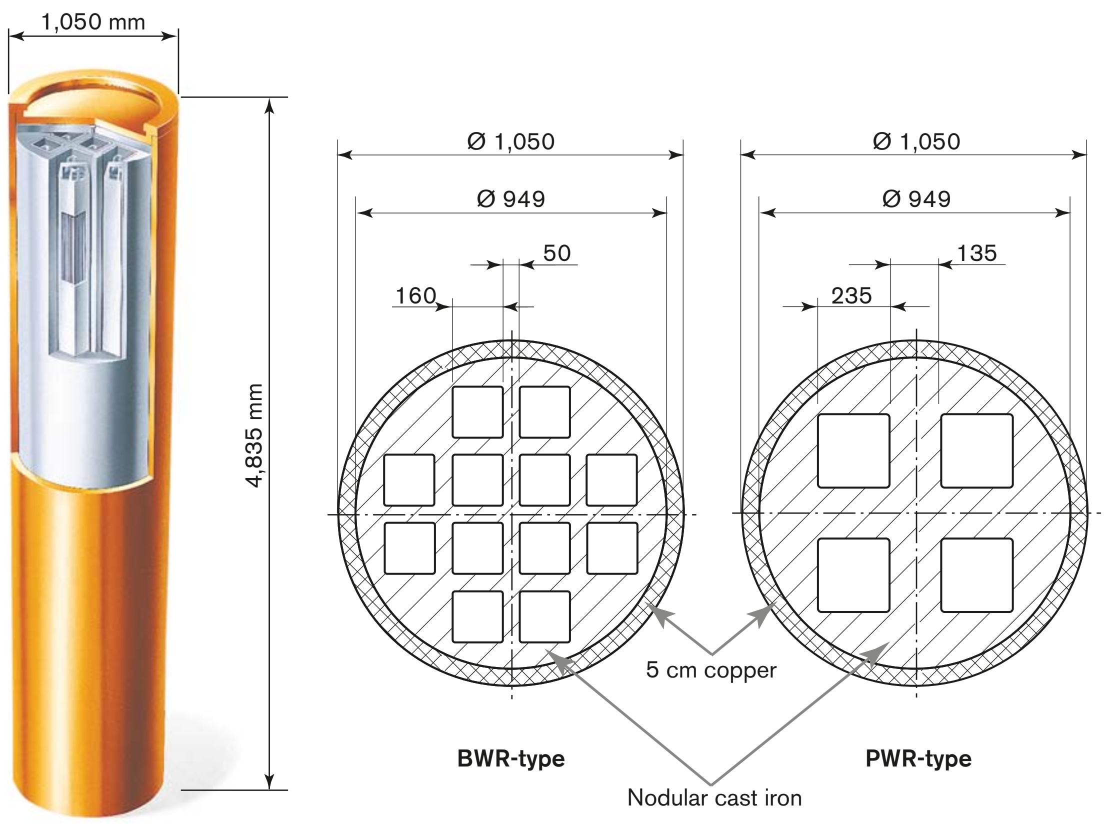


Figure S-5. *Left: The reference design with a corrosion resistant outer copper shell and a load-bearing insert of nodular cast iron. Right: Cross section of insert designs of the BWR and PWR types.*

In a **Canister production report**, it is demonstrated how canisters are to be manufactured and quality assured in order to fulfil the specifications of the reference design. The report also demonstrates that the reference design conforms to the design premises for the canister, by referring to a comprehensive design analysis. It is, therefore, concluded that the reference design together with the suggested production and control methods yield a manufactured canister that conforms to the design premises. One important implication of this is that all the 6,000 canisters are tight at deposition.

Buffer

The main function of the clay buffer is to restrict water flow around the canister. This is achieved by choosing a buffer material with a low hydraulic conductivity after water saturation. This makes diffusion the dominant transport mechanism. The material should also have a sufficient swelling pressure, making the buffer self sealing. The clay material's montmorillonite content is a key property for the safety functions of the buffer.

In SR-Site two example materials that conform to the design premises are assessed. The examples, MX-80 and Ibeco RWC are both from large deposits and are mined by large bentonite suppliers. They are of different origin and should be seen as relevant illustrations of possible alternatives to be used in the repository.

The **Buffer production report** demonstrates how the buffer is to be manufactured and emplaced in a quality assured manner, in order to fulfil the specifications of the reference design.

Deposition tunnel backfill material

The main function of the deposition tunnel backfill is to limit advective transport in the deposition tunnels. This is achieved by choosing a backfill material with a low hydraulic conductivity and a sufficient swelling pressure. The backfill should also contribute to keeping the buffer in place, i.e. it should restrict upwards buffer expansion. This is primarily achieved with a sufficient density of the backfill material.

The reference backfill material is a bentonite clay with the montmorillonite content of 50–60%. In SR-Site one example material, Milos BF 04, that conforms to the design premises is assessed.

The **Backfill production report** demonstrates how the deposition tunnel backfill is to be manufactured and emplaced in a quality assured manner, in order to fulfil the specifications of the reference design.

Additional engineered components in the repository

For the purpose of SR-Site, the additional engineered components in the repository are defined as

1. Deposition tunnel plugs: Presented in the **Backfill production report**.
2. Central area: Presented in the **Closure production report**.
3. Top seal: Presented in the **Closure production report**.
4. Bottom plate in deposition holes: Presented in the **Underground openings production report**.
5. Bore hole seals: Presented in the **Closure production report**.
6. Closure of main tunnels and transport tunnels.
7. Closure of ramp and shafts below the top sealing.
8. Plugs (other than deposition tunnel plugs).

In SR-Site the closure of all tunnels at repository level as well as the ramp and shaft below the top sealing are treated as tunnel backfill, in accordance with the current reference design. All plugs in the repository are treated as deposition tunnel plugs, also in accordance with the current reference design.

The purposes of the closure components are generally to restrict groundwater flow through the underground openings, to provide mechanical restraint and to obstruct unintentional intrusion into the repository. The exception is the bottom plate in the deposition holes which only has the purpose of facilitating the installation of the canister and the buffer.

Underground openings

In all phases of underground design, uncertainties with regard to site conditions must be anticipated. In order to establish a final layout for deposition tunnels and deposition holes, a large volume of rock will have to be characterised, and this characterisation could only effectively be carried out from underground openings. This means that the characterisation will develop as the construction work proceeds.

The depth established for the reference design is a compromise arising from design premises on long-term safety and constructability of the deposition tunnels and deposition holes of the repository facility. Below the depth of 400 m the frequency of water conducting fractures is very low, while the rock stress is still acceptable justifying that the maximum depth of the repository facility is located at elevation −470 metres with a minimum depth (tunnel roof) at elevation −457 metres.

The thermal properties of the site are used to design a minimum spacing of canisters to ensure that the maximum peak temperature in the buffer < 100°C.

The layout is adapted to meet the design premises relating to mitigating earthquake hazard by ensuring that all deposition holes are placed outside the respect distances to deformation zones large enough to potentially host future earthquakes. Furthermore, large fractures are not allowed to intersect deposition positions in accordance with the Extended Full Perimeter Intersection Criterion (EFPC). This criterion requires that a canister position must not be intersected by a fracture that also fully intersects the deposition tunnel perimeter. Furthermore, canister positions that are intersected by fractures that also intersect four or more adjacent positions are rejected.

Potential deposition holes with high inflows are not accepted for deposition. In SR-Site, this is primarily addressed by applying a modified version of the EFPC to avoid deposition positions with potential for high future groundwater flow.

The orientation of the deposition tunnels is related to the orientation of the maximum principal stress in order to mitigate the potential for spalling. Some construction materials in the rock or on rock surfaces, e.g. originating from rock support and from grouting, will remain in the repository after closure.

Summary

In summary, the following are among the most important safety related features of the initial state of the repository:

- The canisters' 5 cm copper shell providing a corrosion barrier.
- The canisters' ability to withstand isostatic loads, provided by the mechanical properties of the cast iron insert.
- The canisters' ability to withstand shear loads, also provided by the mechanical properties of the cast iron insert.
- The deposited buffer density, and the quality assured material composition of the buffer that ensures the development of the buffer into a diffusion barrier when water saturated.
- The deposited density and material composition of the deposition tunnel backfill.
- The general layout of the repository, with respect distances to fracture zones that can potentially host large earthquakes and with a distance between deposition holes that, together with the limitations on thermal output from the deposited canisters, ensure that the temperature of the repository is below 100°C with a sufficient margin.
- Acceptance of deposition positions according to established criteria, which reduces the likelihood that deposition positions are intersected by large and/or highly water conducting fractures.

S3 Analysing safety – the safety assessment

S3.1 Introduction

The repository system will evolve over time. Future states will depend on

- the initial state,
- internal processes, i.e. a number of radiation related, thermal, hydraulic, mechanical, chemical and biological processes acting internally in the repository system over time, and
- external factors acting on the system.

Internal processes are e.g. the decay of radioactive material, leading to the release of heat and the subsequent warming of the fuel, the engineered barriers and the host rock. Groundwater movements and chemical processes affecting the engineered barriers and the composition of groundwater are other examples. External factors include effects of future climate and climate-related processes, such as glaciations and land uplift.

The initial state, the internal processes and the external influences and the way they together determine repository evolution, can never be fully described or understood. There are thus uncertainties of various types associated with all aspects of the repository evolution and hence with the evaluation of safety. A central theme in any safety assessment methodology must therefore be the management of all relevant types of uncertainty. This management amounts to identifying, classifying and describing uncertainties, as well as handling them in a consistent manner in the quantification of the repository evolution and of the radiological consequences to which it leads. A methodological approach also implies comparing the results of the assessment with regulatory criteria in such a way that appropriate allowance is made for the uncertainties associated with the assessment.

The safety assessment SR-Site consists of eleven main steps. Figure S-6 is a graphical illustration of the steps. The methodology followed in the first ten steps of the assessment is described in the following subsections, together with key results from each step. The outcome of the final step, the compilation of conclusions, is described in Section S4.

S3.2 Step 1: Processing of features, events and processes (FEPs)

This step consists of identifying all the factors that need to be included in the analysis. Experience from earlier safety assessments and KBS-3 specific and international databases of relevant features, events and processes (FEPs) influencing long-term safety are utilised. An SKB FEP database is developed where the great majority of FEPs are classified as being either i) initial state FEPs, ii) internal processes or iii) external FEPs. Remaining FEPs are either related to assessment methodology in general or determined to be irrelevant for the KBS-3 concept. Based on the results of the FEP processing, an SR-Site FEP catalogue, containing FEPs to be handled in SR-Site, has been established. The further handling of the three FEP categories is described in the three subsequent steps of the methodology.

This step of FEP processing is fully documented in the SR-Site **FEP report**[1].

S3.3 Step 2: Description of the initial state

The initial state of the system is described, based on the descriptive model of the repository site, the KBS-3 repository design with its different components and a site-specific layout applying this design to the site. The initial state of the geosphere and the biosphere is that of the natural system prior to excavation. The initial state of the fuel and the engineered components is that immediately after deposition.

The initial state of the system is a fundamental input to the assessment and needs thorough substantiation. For the site, this is provided by the site descriptive model of the Forsmark site, in the **Site description Forsmark** report, i.e. the results of the surface based site investigation and the modelling of the site based on the site investigation data. The Forsmark site model is a fundamental reference to SR-Site.

[1] The **FEP report** is one of several principal references in this main report. They are referenced with short-names in bold. The same nomenclature is used in the main text.

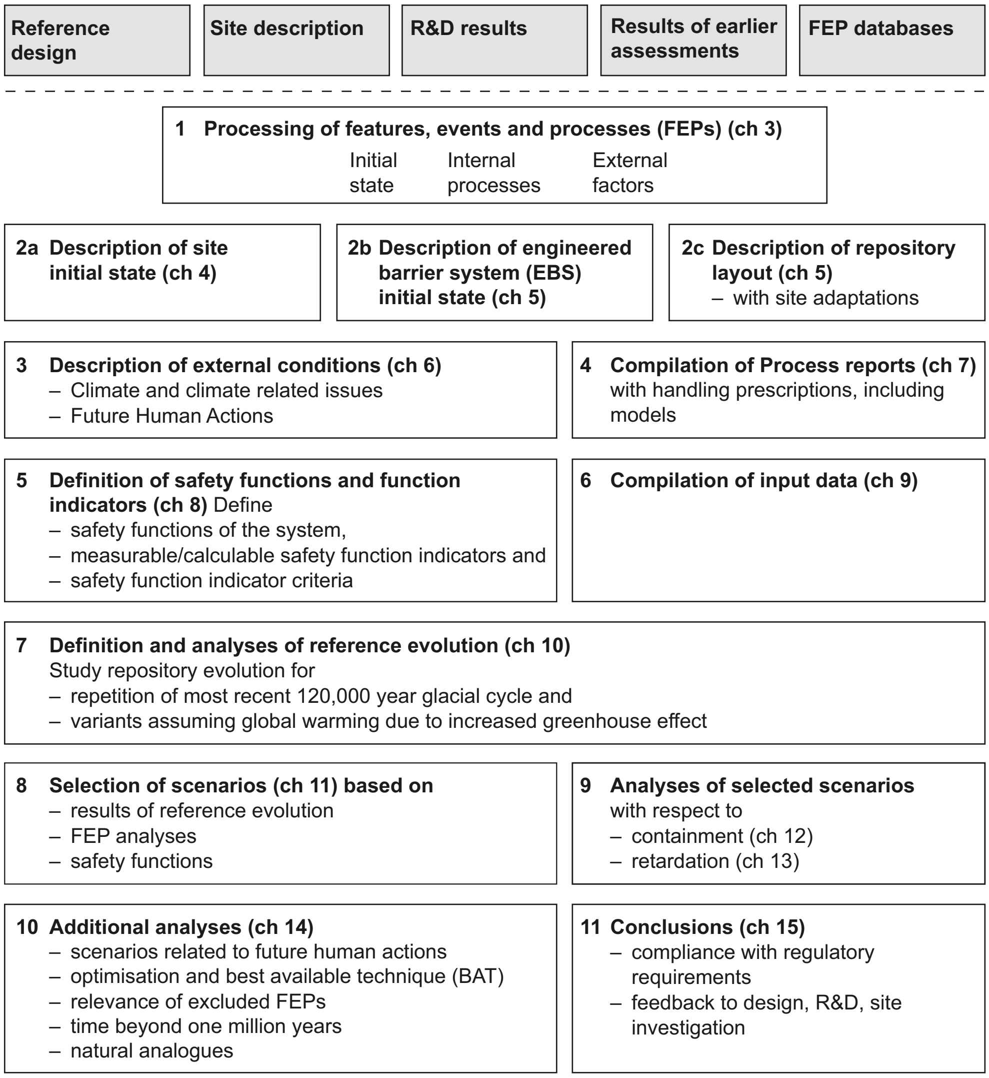


Figure S-6. *An outline of the eleven main steps of the SR-Site safety assessment. The boxes at the top above the dashed line are inputs to the assessment. The chapters in the main report where the steps are further documented are also indicated.*

The initial state of engineered components of the repository system are described in a number of **Production reports** covering the spent fuel, the canister, the buffer, the tunnel backfill, the repository closure and the underground openings constructions, respectively. See further Section S2.3.

S3.4 Step 3: Description of external conditions

Factors related to external conditions are handled in the three categories "climate related issues", "large-scale geological processes and effects" and "future human actions", FHA. The handling of these factors is described in the **Climate report**, the **Geosphere process report**, and the **FHA report**, respectively.

A key point in the handling of external conditions is the establishment of reference external conditions for the subsequent analysis. These reference external conditions postulate a repetition of the last 120,000 year glacial cycle, the Weichselian. An alternative reference evolution is based on the assumption of a global warming effect. In addition, physically possible climate conditions that would have the most severe impact on repository safety are sought for use in the scenario selection in a later step of the assessment.

Future human actions are handled according to a methodology established in the SR-Can assessment with minor updates for SR-Site. Based on a structured account of a large number of FEPs relating to FHA, a selection of stylised cases for further analyses is made.

S3.5 Step 4: Compilation of processes reports

The identification and handling of processes of importance for the long-term evolution and safety of the repository is a key element in the safety assessment. The identification of processes is based on earlier assessments and FEP screening. All identified processes within the system boundary relevant to the long-term evolution of the system are described in three dedicated **Process reports**, one for the fuel and canister, one for the buffer, backfill and repository closure and one for the geosphere. Short-term geosphere processes/alterations due to repository excavation are included in the **Geosphere process report** and are taken into account in the assessment.

Each process is documented in the **Process reports**, following a template with fixed headings. The documentation concludes with establishing how the process will be handled in the safety assessment, constituting the key result from the **Process reports**. The **Process reports** thus provide a "recipe" for the handling of the various processes in the assessment.

The handling of all processes in a process report is summarised in a *process table* describing if the process is neglected, if it is quantitatively modelled or if the choice between neglect and modelling is subject to a specified condition that may or may not be fulfilled as the repository system evolves.

Several of the processes are thus handled through quantitative modelling, where each model in general includes several interacting processes, often occurring in different system parts and hence described in different process reports. The models form a network, where results from one model are used as input to another. The network is described graphically by two *Assessment Model Flowcharts, AMFs*, and two associated *AMF tables* linking the processes in the process tables, the models in the AMFs and the reporting of the modelling exercise in this main report.

S3.6 Step 5: Definition of safety functions, safety function indicators and safety function indicator criteria

A central element in the methodology of the SR-Site assessment is the definition of a set of *safety functions* that the repository system should ideally fulfil over time. Here, the overall safety functions containment and retardation are differentiated into a number of lower level functions for the canister, the buffer, the deposition tunnel backfill and the host rock. The evaluation of the safety functions over time is made possible by associating every safety function with a *safety function indicator*, i.e. a measurable or calculable property of the repository component in question. For several functions, it is also possible to associate a *safety function indicator criterion* such that if the safety function indicator fulfils the criterion, then the safety function in question is upheld.

The ability of the canister to resist isostatic load is an example of a safety function. The associated indicator is the isostatic load on the canister and the criterion is the isostatic load that the canister has been demonstrated to sustain.

It is important to note that the safety function indicator criteria are not the same as design criteria, formally described as design premises in SR-Site. The former should ideally be upheld throughout the assessment period whereas the design premises should be fulfilled initially. In general, the design premises should assure that the system is robust to the extent that the safety functions indicator criteria are fulfilled over time. For example, the copper canister must be designed such that its initial thickness (the design premise) ensures that it will sustain corrosion for a very long time, i.e. such that the thickness is non-zero (the function indicator criterion) during this time.

The set of safety functions in itself provides understanding of the safety features of the system and a list of key issues to evaluate over time in the assessment. The safety functions are explicitly used in later steps of the assessment to evaluate safety in a structured manner over different time frames when analysing a reference evolution. A key use of the safety functions is their role in the selection of a number of scenarios whereby uncertainties related to the safety of the system are evaluated in a structured manner.

All safety functions, safety function indicators and safety function indicator criteria related to containment are summarised in Figure S-7. Each function is briefly discussed in connection with the account of the results of the reference evolution below. Safety functions related to retardation have also been developed and they are summarised in a figure similar to Figure S-7. Many of the criteria

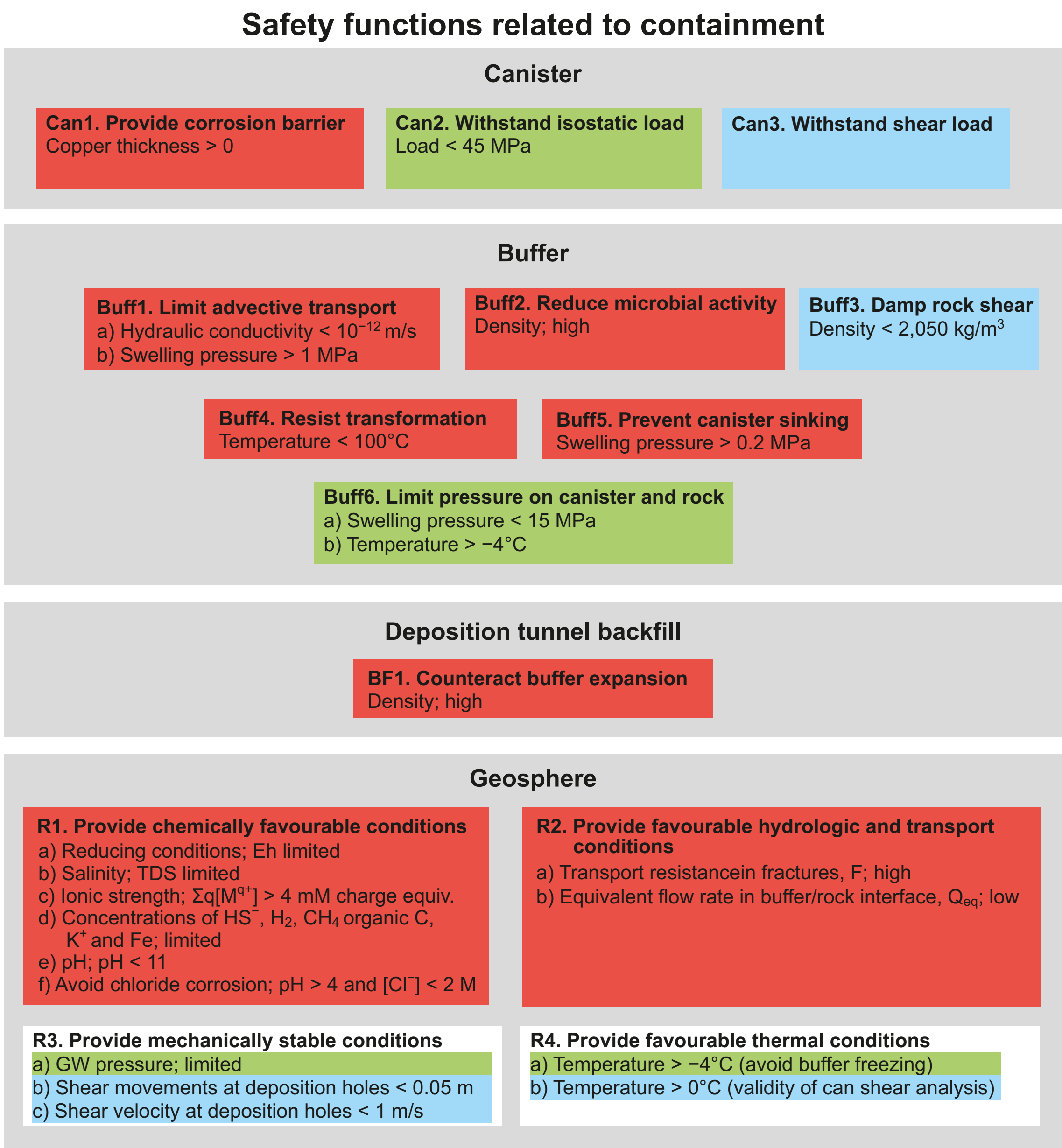


Figure S-7. *Safety functions (bold), safety function indicators and safety function indicator criteria related to containment. When quantitative criteria cannot be given, terms like "high", "low" and "limited" are used to indicate favourable values of the safety function indicators. The colour coding shows how the functions contribute to the canister safety functions Can1 (red), Can2 (green) and Can3 (blue).*

for retardation are related to those for containment. This applies in particular to the functions of the geosphere to provide favourable chemical and hydrologic/transport conditions.

When the safety functions have been defined, a *FEP chart* is developed, showing how initial state factors and processes in the long-term evolution of the repository are related to the safety functions.

S3.7 Step 6: Compilation of input data

In this step, data to be used in the quantification of repository evolution and in dose calculations are selected using a structured procedure. The process of selection and the data values adopted are reported in a dedicated **Data report**. The process follows a template for discussion of input data quality and uncertainties. The instructions concern two parties, the supplier and the customer. The suppliers are the teams providing the data. The customer is in broad terms the SR-Site team that is responsible for performing the SR-Site safety assessment. The models for which data are required are given in the AMF described in step 4 above. The procedures for data auditing and storage form part of the SR-Site quality assurance plan.

S3.8 Step 7: Definition and analyses of reference evolution

In this step, a reference evolution of the repository system that follows from the reference external conditions defined in step 3 is defined and analysed. The purpose is to gain an understanding of the overall evolution of the system and of uncertainties affecting the evolution, for the scenario selection and scenario analyses that follow in the two subsequent steps. The evolution is an important basis for the later definition of a main scenario.

Focus is on the containment capacity of the system. Two cases of the reference evolution are analysed.

1. A base case in which the external conditions during the first 120,000 year glacial cycle are assumed to be similar to those experienced during the most recent cycle, the Weichselian. Thereafter, seven repetitions of that cycle are assumed to cover the entire 1,000,000 year assessment period.
2. A global warming variant in which the future climate and hence external conditions are assumed to be substantially influenced by human-induced greenhouse gas emissions during the first 120,000 year glacial cycle. This analysis is related to that of the base case.

For both these, the initial state with its uncertainties is assumed, all internal processes, with their uncertainties, are handled according to the specification given in the **Process reports**, and data with their uncertainties are taken from the **Data report**.

The presentation of the analysis of the base case of the reference evolution is divided into four time frames:

- The excavation/operational period.
- The first 1,000 years after repository closure and the initial period of temperate domain from the reference glacial cycle.
- The remaining part of the glacial cycle.
- Subsequent glacial cycles up to one million years after repository closure.

For each time frame, issues are presented in the following order:

- Climate issues.
- Biosphere issues.
- Thermal, mechanical, hydraulic and chemical issues in the geosphere.
- Thermal, mechanical, hydraulic and chemical issues for the engineered barrier system (canister, buffer and backfill).

The discussion of each of the issues is concluded with an account of identified uncertainties to be propagated to later stages of the reference evolution and to subsequent parts of the safety assessment.

The commentary on each time frame concludes with a discussion of the expected status of the safety function indicators during and at the end of the time frame.

General development of the reference evolution

A considerable part of the material presented in the reference evolution is results from simulation studies, as established when the handling of processes was determined and as represented graphically in the AMFs.

Initially, the evolution is characterised by a transient caused by the excavation of the host rock and the construction and presence of the repository. In the long term, the evolution is characterised by changes induced by the changing external conditions. Figures S-8 and S-9 show key aspects of the external conditions for the reference evolution, in the form of the modelled reconstruction of the Weichselian glacial cycle.

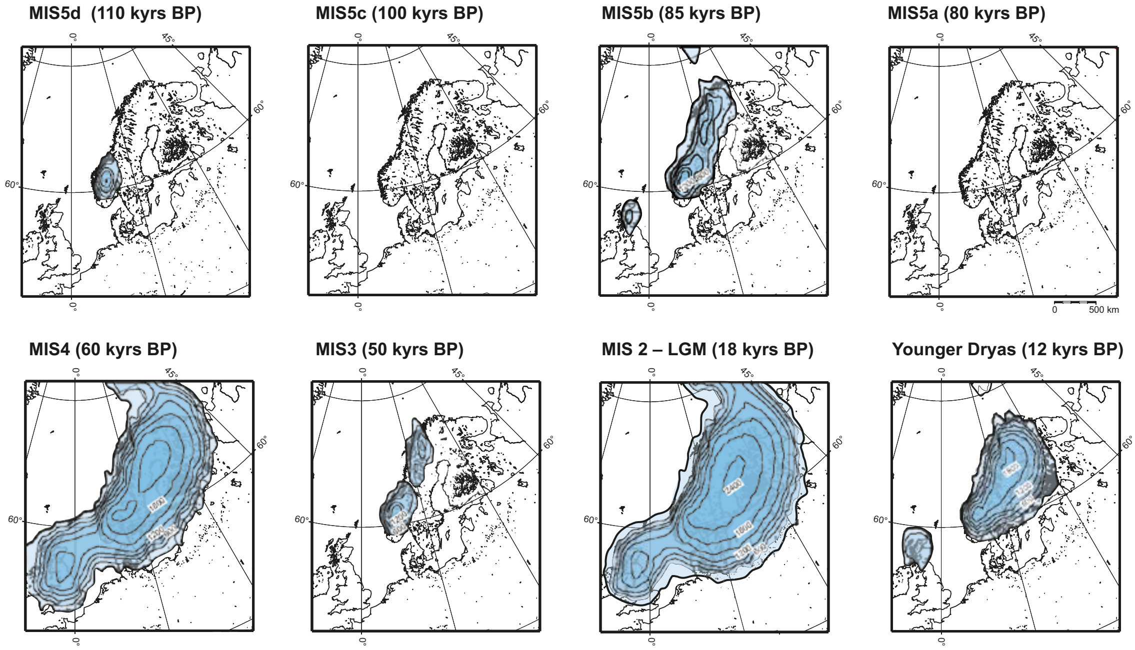


Figure S-8. Selected maps of modelled ice sheet configurations from the reconstruction of the Weichselian ice sheet. Contour lines show ice surface elevation with a 300 m interval. All maps show the present day shore-line position. BP stands for 'before present'.

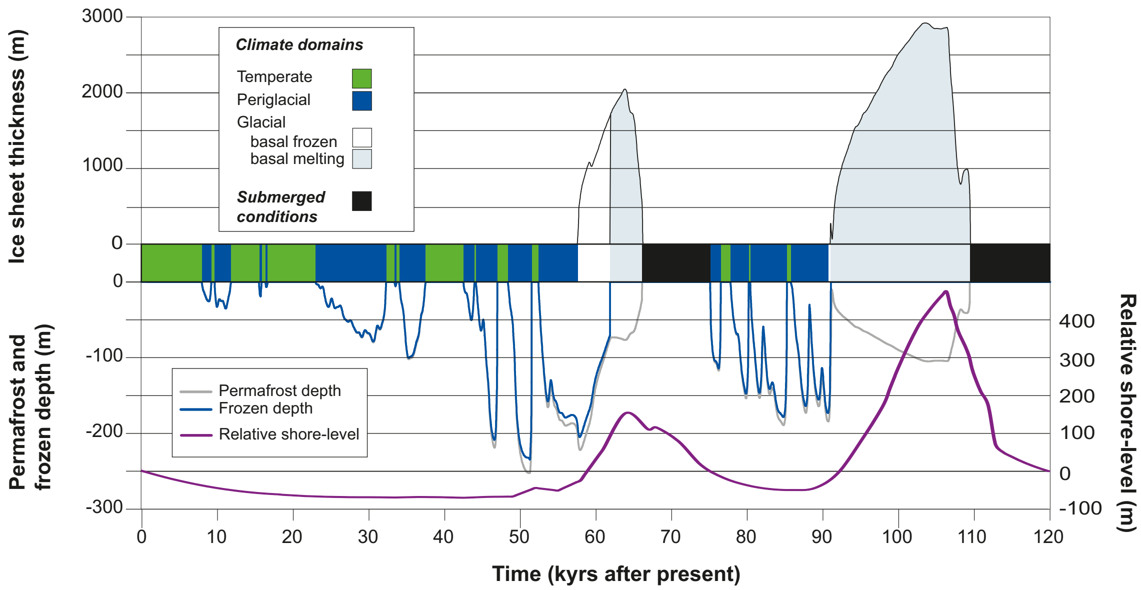


Figure S-9. Reference glacial cycle evolution of important climate-related variables for the Forsmark repository location.

The thermal evolution is characterised by a quick temperature increase due to the heat production in the spent fuel, with peak temperatures in the canister, the buffer and the wall of the deposition hole occurring after a few tens of years. The temperature will return to the background temperature of the host rock (about 11°C at Forsmark) in tens of thousands of years for temperate climate conditions. For glacial and in particular permafrost climate conditions the temperature at repository depth will decrease, but will always be above 0°C.

Mechanically, a damaged zone, in particular below the floors of the deposition tunnels is expected to form during excavation, although it is unlikely that it will form a hydraulically connected pathway, and the initial thermal transients may cause thermally induced spalling of limited extent in the walls of the deposition holes. In the long term, the mechanical conditions are characterised by stability. Large earthquakes in major fracture zones in the vicinity of the repository cannot be excluded, and the layout of the repository is designed to avoid canister failures in such events.

Hydraulically, the repository is drained during construction and operation. The times required for water saturation of the deposition tunnel backfill and of the buffer will vary considerably between different parts of the repository and are likely to range from a few tens of years to several thousand years, as a consequence of the rock properties at Forsmark. In the long term, the flow conditions in the rock are controlled by changing external conditions.

Chemically, initial transients are expected. These include the consumption of initially entrapped oxygen in the repository by microbes and minerals in the rock, in the backfill and the buffer, and reactions involving construction and stray materials in the repository. The transients are followed by slowly varying conditions during the initial temperate stage and changes controlled by changing external conditions in the long term.

Safety and safety functions during the reference evolution

For the majority of the 6,000 deposition positions, all safety functions relating to the canister, the buffer, the deposition tunnel and the host rock are assessed to be satisfactorily upheld during the reference evolution. The following results and conclusions regarding safety functions emerged from the analysis of the reference evolution. The nomenclature for the safety functions is that used in Figure S-7.

The rock

The results from the studies of the *hydrogeochemical development* over the reference evolution imply the following.

- Reducing conditions (safety function R1a) are upheld throughout the reference evolution, once they have been established shortly after repository closure. Local, temporary penetration of oxygen to repository depth cannot be excluded during hydrologic transients caused by the passage of an ice sheet margin but the potential effects are too small to influence safety.
- The salinity (R1b) is limited to below 0.35 M in terms of chloride, i.e. below concentrations where the function of the buffer or backfill could be impaired. The highest salinities occur as a result of upconing i) during the initial geochemical transient caused by the draining and subsequent resaturation of the rock and ii) during transients caused by the passage of an ice sheet margin above the repository.
- The ionic strength (R1c) is above 4 mM charge equivalents, required for exclusion of buffer colloid release/erosion, for the majority of deposition positions throughout the reference evolution. However, it cannot be ruled out that a few percent of the deposition positions, those intersected by fractures with the highest flow rates, are exposed to dilute waters such that buffer erosion could occur after thousands of years of temperate or glacial conditions or during short transients caused by the passage of an ice sheet margin above the repository.
- Concentrations of agents detrimental for the buffer and canister are (R1d) as follows.
  - Concentrations of HS^- are expected to not exceed present day concentrations.
  - Concentrations of H_2 are expected to remain below 0.1 mM. Hydrogen produced by the corrosion of steel and iron repository components is expected to either diffuse away or be used in microbial processes. If sulphate reduction is involved, the sulphide produced is expected to react with the iron(II) from the corrosion and increased sulphide levels will not occur due to this mechanism.
  - Concentrations of CH_4 and organic C will remain below ~0.1 mM and ~1 mM, respectively.
  - Concentrations of K^+ and Fe will remain below ~5 mM and 0.1 mM, respectively.
- The pH of the groundwater (R1e) is below 11 and above 6.3.
- The conditions required to avoid chloride assisted corrosion (R1f) are satisfied given the limited chloride concentration (<0.35 M) and the pH values of the groundwaters (>6.3).

The results from the studies of the *hydrogeological development* over the reference evolution imply the following.

- The equivalent flow rates in the buffer/rock interface (R2b) are low. Almost 5,000 of the 6,000 canister positions are not intersected by water bearing fractures according to the distribution of fracture sizes and intensities in the hydrogeological model. Among the intersected fractures, most have a very low equivalent flow rate. However, the span in flow rates is wide when the entire ensemble of deposition positions is considered and the highest percentile of the flow rate distribution is such that they can contribute to long-term detrimental effects on the buffer and the canister.
- The integrated transport resistance in the network of fractures connecting deposition positions to the surface (R2a) is high for most deposition positions, contributing to efficient retention of radionuclides in the rock. However, the span in transport resistances is wide when the entire ensemble of deposition positions is considered and the lowest percentile of distribution is such that retention in the rock may be poor. Furthermore, transport paths with low transport resistances often have a high flow rate at the deposition position.

Regarding the *mechanical stability* of the host rock, the following is concluded.

- The groundwater pressure (R3a) is usually in the order of 5 MPa, but may increase if the site is covered by ice. During glacial conditions the pressure will depend on the height of the ice cover, and will stay below 26 MPa for the modelled reference evolution.
- Shear movements at deposition holes (R3b) exceeding 5 cm are extremely rare, due to the low probability of occurrence of large magnitude earthquakes in the vicinity of the repository, due to the use of respect distances to major fracture zones in selecting deposition areas and in the application of acceptance criteria in the selection of individual deposition positions.
- Shear velocities at deposition holes (R3c) are, for the same reasons as above, below 1 m/s.

Regarding the *thermal development* of the rock, the temperature at repository depth at Forsmark is above 0°C with a large margin in the reference evolution, i.e. buffer freezing is avoided (R4a) and the shear analyses of the canister are valid (R4b).

The deposition tunnel backfill

Regarding the deposition tunnel backfill, the density is sufficient to counteract buffer expansion (BF1) for all cases analysed in the reference evolution.

The buffer

In the majority of deposition holes, the buffer is expected to maintain a density over time such that, given the initial state properties of the buffer, the favourable composition of the groundwater over time and the limited peak temperatures achieved by an appropriate layout of the repository, all its safety functions are fulfilled, i.e. so that:

- Advective transport is limited through a hydraulic conductivity below 10^{-12} m/s (Buff1a) and a swelling pressure above 1 MPa (Buff1b).
- Microbial activity is suppressed (Buff2).
- Rock shear is damped (Buff3), by ensuring an upper limit on the density.
- Buffer transformation is avoided (Buff4) through an upper limit on the temperature.
- Canister sinking (Buff5) is avoided through a swelling pressure above 0.2 MPa.
- The pressure on the canister and the host rock is limited (Buff6) through a swelling pressure below 15 MPa and a rock temperature above –4°C.

However, as a sufficiently high ionic strength (R1c) is not ensured in the long term for the deposition positions intersected by the fractures with the highest flow rates, it cannot be ruled out that the buffer in these positions will be eroded. The calculated erosion rates are such that advective conditions could arise in around one percent of the deposition holes during the one million year assessment period. For these holes, the safety functions Buff1 and Buff2 are violated.

The canister

Regarding the canister's role as a corrosion barrier (Can1), the analyses in the reference evolution show that at most a few mm of the 5 cm thick copper shell will be corroded in one million years if the buffer is in place with its safety functions maintained. However, for the few deposition positions where the buffer may be lost due to erosion such that advective conditions arise in the deposition hole, the corrosion rate is enhanced. The quantitative analyses show that, as a statistical average, up to around one canister may have failed due to corrosion under such circumstances at the end of the one million year assessment period.

Regarding the canister's ability to withstand isostatic loads (Can2), the quantitative assessments show that the total isostatic pressure is below the 45 MPa design premise of the canister, meaning that this safety function of the canister is maintained.

Regarding the canister's ability to withstand shear loads, the analysis shows that the probability of one such failure having occurred at the end of the one million year assessment periods in the ensemble of 6,000 canisters is 0.08, with a number of pessimistic assumptions regarding both the host rock and the canister.

Uncertainties identified in the analysis of the reference evolution

The uncertain issues that need to be propagated to scenario analyses are essentially connected within two groups: issues related to canister failure due to corrosion (safety function Can1) and issues related to canister failure due to shear loads (safety function Can3), whereas canister failures due to isostatic load (Can2) are ruled out according to the assessments in the reference evolution.

The issues relating to canister failure due to corrosion are as follows:

- Groundwater flow over the glacial cycle.
- Groundwater salinity over the glacial cycle.
- Buffer erosion, determined by groundwater flow, fracture apertures and salinity, and the assessment of which is also affected by the incomplete conceptual understanding of buffer erosion.
- Groundwater sulphide concentrations over the glacial cycle.
- Canister corrosion under advective conditions, requiring buffer erosion to the extent that advective conditions arise in the deposition hole, and then determined by groundwater flow and sulphide concentrations.

The issues relating to canister failure due to shear load are as follows:

- The occurrence of earthquakes of a sufficient magnitude to cause secondary shear movements in fractures intersecting deposition holes.
- The extent of detrimental secondary shear movements given sufficiently large earthquakes.
- The impact of secondary shear movements on the buffer/canister system.

By definition, the external conditions for the reference evolution are constrained either to a development compatible with a repetition of the Weichselian glacial cycle (the base case) or to one compatible with the global warming variant. There are uncertainties within these constraints, leading to uncertainties within the reference evolution, as listed above. There are also significant uncertainties due to the fact that external conditions other than those defining the Weichselian base case or the global warming variant can be conceived. The latter are handled in the scenario analyses.

S3.9 Step 8: Selection of scenarios

Method for scenario selection

A key feature in managing uncertainties in the future evolution of the repository system is the reduction of the number of possible evolutions to analyse by selecting a set of representative scenarios. The selection focuses on addressing the safety relevant aspects of the evolution expressed at a high level by the safety functions 'containment' and 'retardation' which are further characterised by reference to safety function indicators.

The selected scenarios should cover all reasonable future evolutions. Furthermore, it should be possible to logically calculate the risk associated with the presence of the repository as a sum of risk contributions from the set of scenarios.

There are also several issues concerning applicable regulations that have to be taken into account in the selection of scenarios. Given the regulatory requirements and the general considerations above, a method for the selection of scenarios in five steps has been developed as explained below.

1. Definition of the main scenario

A *main scenario* is defined, based on the reference evolution and in accordance with SSMFS 2008:21. The main scenario is split into two variants, based on the two variants of the reference evolution, (the Weichselian base case and the global warming variant).

2. Selection of additional scenarios based on potential loss of safety functions

A main factor governing scenario selection is the concern that the safety functions relating to containment should be upheld. Therefore, these safety functions are used to structure the selection of additional scenarios. This is the main approach for addressing the issue of *less probable scenarios*, in SSMFS 2008:21.

There are three canister safety functions related to containment: to provide a corrosion barrier, to withstand isostatic load and to withstand shear load. Three distinct canister failure modes, due to corrosion, isostatic pressure and shear movement, respectively, can thus be derived from the safety functions. Therefore, three scenarios, one for each canister failure mode, are generated. Three 'failed' states of the buffer; advective, frozen and transformed, are also considered as scenarios. The canister scenarios are systematically combined with the buffer scenarios.

For each selected scenario, uncertainties related to initial state factors, processes and external conditions that are not covered in the main scenario are considered. In e.g. the case of canister failure due to isostatic overpressure, inadequacies in the manufacturing of the load-bearing canister insert, higher than reference buffer swelling pressures and extreme ice sheets yielding high groundwater pressures are considered.

An assessment of whether each scenario is to be considered as "less probable" or "residual" is made. In the former case, the likelihood of the scenario is normally pessimistically set to one, whereas the assessed limited likelihoods of its characteristic FEPs, e.g. large earthquakes, are taken into account in the risk calculation associated with the scenario.

These scenarios also cover many of the residual scenarios required by SSM's Regulations and General Guidance to analyse *the significance of barriers and barrier functions*. To obtain a deeper understanding of barrier functions, a number of residual scenarios are defined illustrating, from the point-of-view of radionuclide transport, hypothetical situations where one or several barriers are assumed to be initially lost.

3. Scenarios related to future human actions

A set of scenarios related to future human actions was also defined and analysed. Human intrusion scenarios resulting in a degradation of system performance are to be considered as "less probable scenarios" according to SSMFS 2008:21, but not included in the risk summation according to the General Guidance to SSMFS 2008:37. SSM requires residual scenarios to illustrate *damage to humans intruding into the repository* and cases to illustrate the consequences of an *unclosed repository that is not monitored*.

4. Other residual scenarios, etc.

Any other scenarios that are, for any reason, considered necessary in order to obtain an adequate set of scenarios are also to be defined. These could include scenarios directly identified in the FEP analysis but not according to the criteria above. No such issues have been identified in SR-Site.

5. Combination of scenarios

For the scenario selection to be comprehensive, combinations of the scenarios must be considered. This is done when all the scenarios have been selected and analysed. Related to the issue of combination of scenarios is that of different event sequences. The sequence in which different events or aspects of the evolution occur may be important for the evolution of the repository. This is explicitly addressed within each scenario.

Summary

In summary, the scenario methodology is an investigation of all routes to the three identified canister failure modes aiming at ruling them out or at quantifying them, considering all conceivable evolutions of the system. The safety functions of the repository components and the understanding of the development of the repository system emerging from the analysis of the reference evolution form the basis for exhaustive evaluations of such routes.

Selected scenarios

The following scenarios are selected in SR-Site.

- A main scenario, corresponding to the reference evolution.
- A buffer advection scenario exploring the routes to and quantitative extent of advective conditions in the deposition hole.
- A buffer freezing scenario exploring the routes to buffer freezing.
- A buffer transformation scenario exploring the routes to buffer transformation.
- A scenario exploring the routes to and quantitative extent of canister failures due to corrosion.
- A scenario exploring the routes to and quantitative extent of canister failures due to shear load.
- A scenario exploring the routes to canister failures due to isostatic load.
- Hypothetical, residual scenarios to illustrate barrier functions.
- Scenarios related to future human actions.

S3.10 Step 9, part 1: Analysis of containment potential for the selected scenarios

Method

The analysis of the selected scenarios is divided in two steps: analysis of containment potential and of retardation potential.

The containment potential is not further analysed in the main scenario; the results from the reference evolution in step 7 are adopted.

The additional scenarios are analysed by focussing on the factors potentially leading to situations in which the safety function in question is not maintained. In most cases, these analyses are carried out by comparison with the evolution for the main scenario, meaning that they only encompass aspects of repository evolution for which the scenario in question differs from the main scenario.

A common template, with a set of fixed headings, is followed in the analysis and documentation of the additional scenarios.

Results

In summary, the following conclusions were reached when the containment potential of the selected scenarios was analysed. (For the hypothetical residual scenarios containment failures are postulated and FHA scenarios are analysed with a different methodology.)

- Buffer advection: This situation may occur in the reference evolution. The additional analyses within the buffer advection scenario, considering conceptual uncertainties and additional interpretations of the hydraulic properties of the sites, suggest a range in the possible extent of buffer advection. *These consequences were propagated to the canister corrosion scenario.*

- Buffer freezing: Buffer freezing was ruled out in the reference evolution, even for an eroded buffer. The additional analyses within the buffer freezing scenario also led to the conclusion that freezing of an intact buffer is ruled out and hence should be considered as a residual scenario. This applies also to the freezing of water in cavities of a partially eroded buffer. *The possibility of buffer freezing was, therefore, not propagated to the canister scenarios.*
- Buffer transformation: The analyses of a high buffer temperature, or other circumstances leading to the transformation of the buffer material within the buffer transformation scenario led to the conclusion that this should be considered as a residual scenario. *The possibility of buffer transformation was, therefore, not propagated to the canister scenarios.*
- Canister failure due to corrosion: This failure mode is included in the reference evolution, where it occurs for the case of advective transport through an eroded buffer and with sulphide in the groundwater as the important corroding agent. In the canister corrosion scenario, all mechanisms for canister corrosion were revisited. It was confirmed that chloride assisted copper corrosion, requiring high chloride concentrations and very low pH compared to what can be expected in the repository, can be ruled out. It was also shown that a recently suggested mechanism for copper corrosion in pure water, for which the scientific basis is judged weak, has a negligible impact on the overall extent of copper corrosion under the assumption that the mechanism exists. The additional analyses in the canister corrosion scenario, with input from the buffer advection scenario, led to the conclusion that advective conditions in the buffer is indeed the main potential cause of corrosion failures. It was also concluded that sulphide in the groundwater is the only corroding agent that has the potential to cause canister failures and then only under advective conditions. Evaluating all the advective situations and other uncertainties related to corrosion led to a range of potential extents of corrosion failure. *These are propagated to the analysis of consequences for the corrosion scenario.*
- Canister failure due to isostatic load: This failure mode was ruled out in the reference evolution and the analysis in the isostatic load scenario led to the conclusion that it should be considered as a residual scenario. Consequences for a hypothetical case of canister failure due to isostatic load are nevertheless analysed.
- Canister failure due to shear load: This failure mode was analysed in the reference evolution, where it had a low probability of occurrence even with a number of pessimistic assumptions. This conclusion remains after the additional analyses in the shear load scenario. *The pessimistically estimated frequency of canister failures due to shear load is propagated to the analyses of consequences for the shear load scenario.*
- The analysis of combinations led to the conclusion that relevant combinations and gradual developments of phenomena either have been addressed in earlier parts of the assessment and in some cases propagated to consequence calculations, or can, with relatively simple complementary arguments, be shown not to give rise to additional cases for further consideration. The only exception concerns a case where a shear failure is followed by buffer erosion. This case was also propagated to consequence calculations.

S3.11 Step 9, part 2: Analysis of the retardation potential for the selected scenarios

This second step of the scenario analyses encompasses calculations of radionuclide release, transport and dose impacts for potential failure modes of canisters identified for each scenario in the analysis of containment potential. The purpose is to assess the retardation properties of the system for these scenarios and to quantify risk. Full documentation of the consequence calculations are provided in the SR-Site **Radionuclide transport report**.

Criticality

If a canister failure occurs, the issue of nuclear criticality has to be considered, since, if this occurred, it could have a strong influence on the further development of the failed canister. The conclusion of criticality analyses in SR-Site is that the canister remains subcritical in the repository for all reasonably conceivable cases.

Biosphere

Over the time scales of relevance for the safety assessment, the biosphere will undergo considerable changes, in particular due to the long-term climatic variation involving glaciations cycles and the associated displacement of the shoreline. In the case of releases of radionuclides from the repository, the potential activity concentrations in surface water are expected to increase as the site emerges from the sea and radionuclides which may have accumulated in sediments can enter into the terrestrial food web in existing or agriculturally converted wetlands. Thus, the potentially highest exposure of humans and other organisms to radionuclides from the repository is expected when at least parts of the site have emerged from the sea. In SR-Site, the development of the landscape at the Forsmark site is modelled in detail for an interglacial climate period. The landscape is divided into a number of objects, for each of which the detailed development over time is modelled. Unit release rates of relevant nuclides are then fed to each object and the resulting time dependent turnover of radionuclides is determined. Each object is pessimistically assumed to be populated by humans to the extent that it can just wholly feed its inhabitants, meaning that they will eat only contaminated food. Humans are assumed to cover their drinking water demand by equal contributions from a contaminated well, and from the surface water in the lake or stream passing through the object. The time dependent radionuclide distribution in each object leads to annual doses to its inhabitants that vary in time. For each nuclide, the peak annual dose over the entire interglacial and over all objects is determined, yielding a peak annual dose per unit radionuclide release rate to the landscape. These landscape dose conversion factors (LDFs) are used in converting release rates from the repository to annual doses. The approach relies on the fact that releases from the repository in general vary slowly over the time scale of an interglacial climate episode.

Results of radionuclide transport and dose for risk contributing scenarios

There are two scenarios for which canister failures cannot be excluded according to the analysis of the containment potential; the corrosion scenario and the shear load scenario.

For the corrosion scenario, a set of cases covering the range of potential extents of corrosion failure from the analysis of the containment potential is considered. The calculated mean doses are at least one order of magnitude below the dose corresponding to the regulatory risk limit, see Figure S-10. In the most pessimistic variants of this scenario, the first canister failures and hence the first releases occur after around 50,000 years. In these variants, the mean dose is about two orders of magnitude below the regulatory limit at 100,000 years and about one order of magnitude below the limit at one million years.

Different extents of canister failures in the corrosion scenario propagated from the analysis of containment potential led to variations in calculated mean doses within one order of magnitude. Also uncertainties in the conceptualisation of the near field transport conditions have a similarly limited impact on the calculation results.

For the shear load scenario, the pessimistically derived frequencies of canister failures due to shear load from the analysis of containment potential are used in the consequence calculations. The calculated mean dose for the initial 1,000 years is negligible in comparison to the dose corresponding to the regulatory risk limit. Between 1,000 and 100,000 years, the calculated mean dose is about three orders of magnitude below the limit and then increases to become about two orders of magnitude below the limit at one million years, se Figure S-11.

Sensitivity analyses of the probabilistic calculation results show that input uncertainties for the fuel dissolution rate, the failure time of the canister and the flow related transport resistance in the geosphere account for most of the uncertainty of the calculated dose. Additional uncertainties are addressed through the formulation of variant calculation cases regarding e.g. different conceptual hydrogeological models or through pessimistic assumptions regarding, e.g. the likelihood of canister failure due to shear load. The influences of all important uncertainties on the calculated risk are evaluated based on the outcome of the risk summation, and possibilities of reducing the uncertainties are discussed as feedback from the assessment.

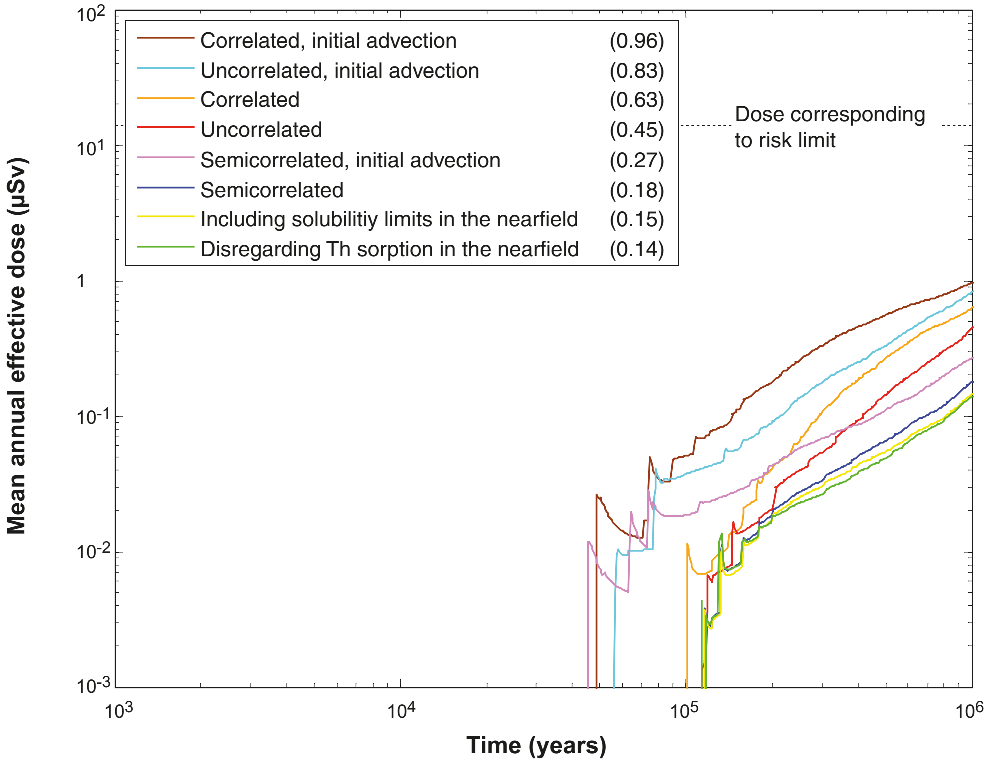


Figure S-10. Summary of far-field mean annual effective dose for all probabilistic calculations performed for the corrosion scenario. The peak doses are given in parentheses in µSv. In the legend, 'Correlated', 'Uncorrelated' and 'Semicorrelated' refer to three variants of the hydrogeological model used in SR-Site. The three 'initial advection' cases put upper bounds on the possible consequences of buffer erosion.

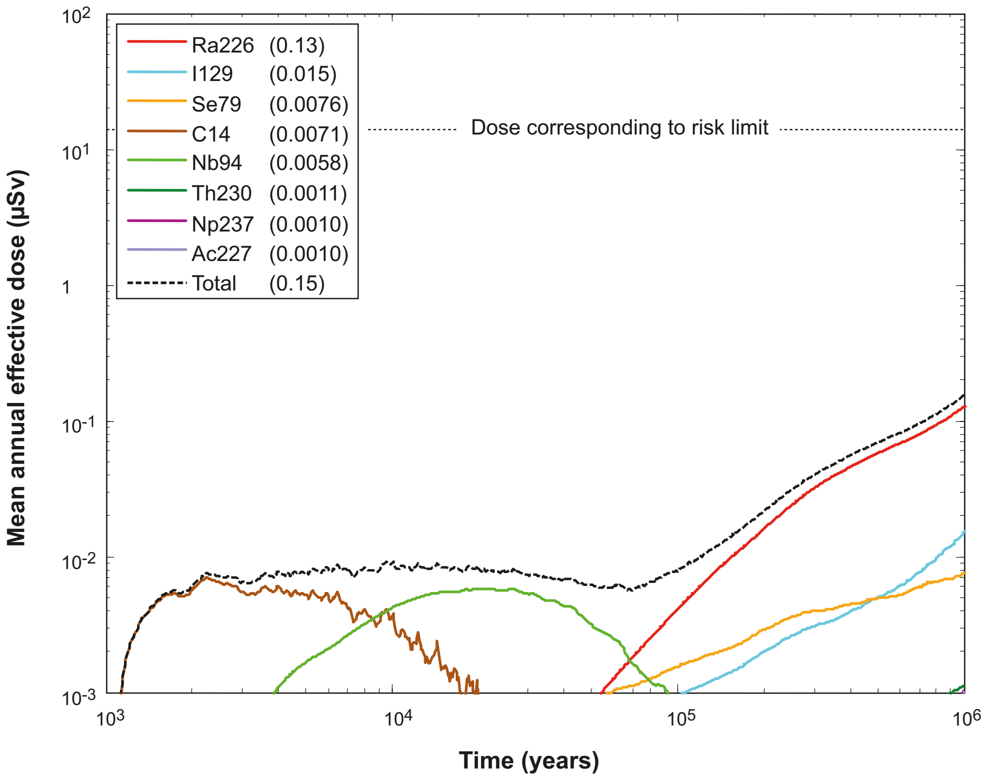


Figure S-11. Probabilistically calculated consequences of shear failure, for the period between 1,000 years and one million years. The legends are sorted according to descending peak mean annual effective dose over one million years (given in brackets in µSv).

Calculation cases with hypothetical complete loss of barrier functions

To illustrate the role of the barriers and as a basis for the discussion of the safety concept, a set of calculation cases with hypothetical, complete losses of barrier functions have also been analysed. The following cases of barrier deficiencies are postulated:

A. An initial absence of enough buffer to cause advective conditions in the deposition hole for all 6,000 deposition holes.

B. An initial pinhole in the copper shell for all 6,000 canisters.

C. An initial, large opening in the copper shell and in the cast iron insert for all 6,000 canisters.

D. A combination of cases A and C, i.e. an initial large opening in all canisters and advective conditions due to loss of buffer for all 6,000 deposition holes.

E. A combination of case C with an assumption of fast fuel dissolution and fast corrosion of metal parts. An initial, large opening in every canister is combined with the assumption of a complete fuel dissolution and metal corrosion in only 100 years.

A loss of the radionuclide retention capability of the rock is combined with each of the five cases, yielding a total of ten release situations. The cases without geosphere retention are denoted A* through E*.

In all cases it is assumed that the backfill and closure are installed and perform as expected. Also, all aspects of the rock other than those related to retention, e.g. the near-field groundwater flow, which is generally low and with only about one sixth of the deposition holes connected to a water conducting fractures, as well as the stable and favourable groundwater composition in the near-field, are assumed to be present. Elemental solubilities are imposed on concentrations of radionuclides in the canister void volume only if the buffer is in place. This is the same approach as used in the analyses of the corrosion and shear load scenarios. The summed dose for each case is given in Figure S-12.

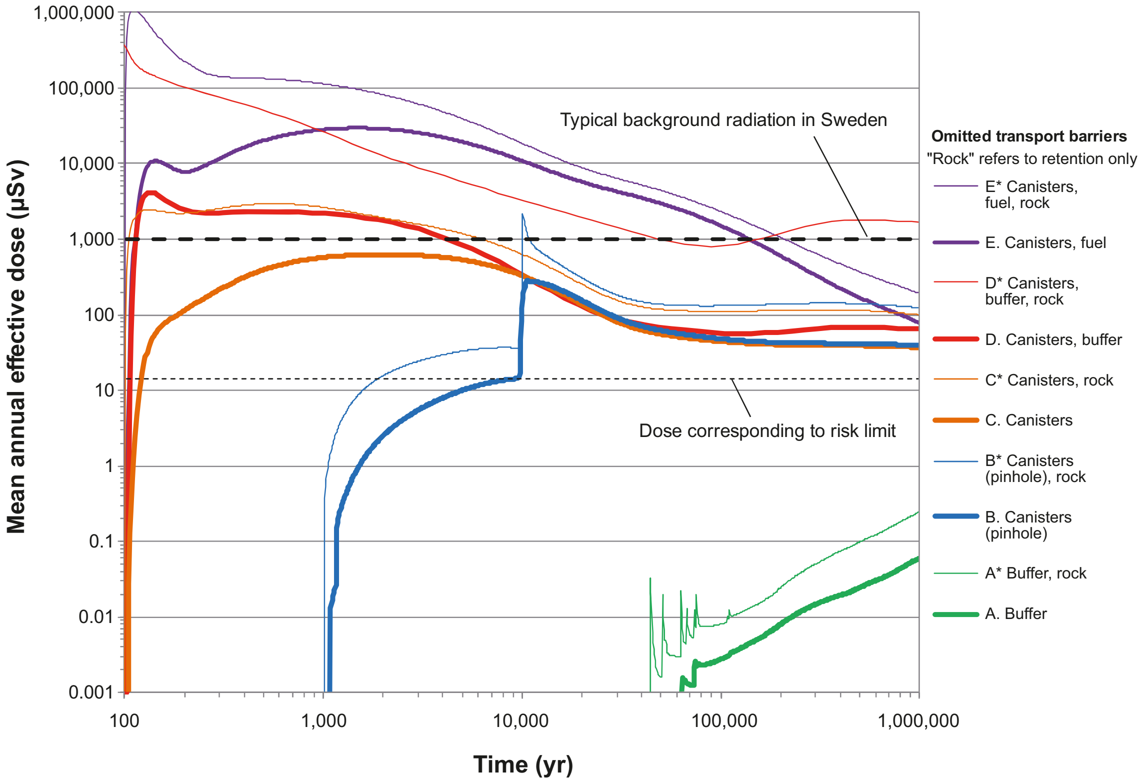


Figure S-12. *Results of stylised cases to illustrate loss of barrier functions. Note that an omission of the "rock" barrier in these cases refers to omission of retention of radionuclides in the rock fractures only, whereas the favourable, low flow rate at repository depth and the favourable geochemical conditions are still taken into account.*

After about 10,000 years, the doses for all cases are below the dose caused by typical background radiation in Sweden, except the case where retention properties of canisters, buffer and rock are all disregarded (case D*) and that with rapid conversion of the waste form in combination with failed canisters (cases E and E*). The low flow and favourable groundwater chemistry of the rock and the presence of backfill and closure of repository tunnel thus provide substantial protection from a waste with unaltered conversion rate.

Additional results

Further results from the radionuclide and dose calculations are presented in the conclusions section below. These include a summation of risk contributions from scenarios which could not be ruled out from the assessment, the calculations of doses to non-human biota, calculations with alternative, simplified analytical models, and the use of alternative safety indicators.

S3.12 Step 10: Additional analyses and supporting arguments

Overview

In this step of the assessment, a number of additional analyses, required to complete the safety assessment, are carried out.

- Analyses of scenarios related to future human actions.
- Analyses required to demonstrate optimisation and use of best available technique, BAT.
- Verification that FEPs omitted in earlier parts of the assessment are of negligible significance in the light of the completed scenario and risk analysis.
- A brief account of the time period beyond one million years.
- Use of natural analogues.

Only results from the first point are given below. Results of the remaining points are summarised as appropriate in the conclusion section of this summary.

Scenarios related to future human actions

Based on generally accepted principles and SSM's Regulations, the future human actions considered are restricted to global pollution and actions that are carried out after the sealing of the repository, take place at or close to the repository site, are unintentional, and impair the safety functions of the repository's barriers. A systematic approach including a technical analysis, an analysis of societal factors, a choice of representative cases and, finally, scenario descriptions and consequence analysis of the chosen cases is adopted. The main conclusions of these analyses are summarised below.

- For a stylised case where a drilling team unintentionally penetrates a canister and part of spent fuel is brought to the surface, the following is found.
  - The dose rate that a member of the drilling personnel would be exposed to while working in the highly contaminated area can be quite high. However, if the drilling occurs at c. 5,000 years after repository closure, the dose rate will have decreased to a value below 1 mSv/hour.
  - The total dose from using the borehole in the drilling case as a well 300 years after repository closure is below background radiation.
  - The maximum total annual effective dose from agricultural use of soil contaminated by fuel debris is very high, but it is noted that there are a number of simplified, pessimistic assumptions made in the calculations.
- The impacts of an open investigation borehole on the groundwater flow and on the long-term properties of the backfill in the deposition tunnel in the vicinity of the borehole are assessed as negligible.
- A tunnel constructed in the upper part of the bedrock would not affect the groundwater flow at repository depth such that the presence of the tunnel violates the safety functions of the deep repository.
- Exploitation of the potential mineral resources in the vicinity of the Forsmark site would not impact the safety functions of the repository.

- Abandoning the repository without backfilling and sealing all parts of the repository may, according to the analysis of a stylised FHA scenario, imply that backfill in the deposition tunnels is lost and that the safety functions for containment are violated for deposition holes located close to the entrance of the deposition tunnels. Without backfill in parts of the system, no canister failures are expected during the first period of temperate conditions. During the subsequent glacial period assumed to last until 66,000 years after present, corrosion breakthrough may occur and the calculated annual effective dose from radionuclides in the failed canisters exceeds the regulatory risk limit. Considering the large uncertainties and cautious assumptions made in the analysis, the result is seen as a simplified illustration of possible consequences, pointing to the necessity of properly backfilling and sealing the repository.

S4 Conclusions of the SR-Site assessment

As mentioned initially, the central conclusion of the safety assessment SR-Site is that a KBS-3 repository that fulfils long-term safety requirements can be built at the Forsmark site.

This conclusion is reached since the favourable properties of the Forsmark site ensure the required long-term durability of the barriers of the KBS-3 repository. In particular, the copper canisters with their cast iron insert have been demonstrated to provide a sufficient resistance to the mechanical and chemical loads to which they may be subjected in the repository environment.

The detailed analyses, performed systematically according to a well-defined methodology, demonstrate that canister failures in a one million year perspective are rare. Even with a number of pessimistic assumptions regarding detrimental phenomena affecting the buffer and the canister, they would be sufficiently rare that their cautiously modelled radiological consequences are well below one percent of the natural background radiation.

S4.1 Overview of results

Compliance with regulatory risk criterion

A repository at Forsmark is assessed to comply with the regulatory risk criterion

The analyses carried out in SR-Site show that a KBS-3 repository at Forsmark constructed in accordance with the current reference design will comply with the regulatory risk criterion issued by SSM.

The likelihood of canister failures during the initial one thousand years, is assessed as negligible

The pessimistically calculated mean number of canisters failing due to earthquakes during the initial one thousand years is of the order of one in a hundred thousand. All other failure types are assessed as ruled out for this period. Furthermore, the evaluations of the canister sealing procedure, have led to the conclusion that all canisters will be tight at deposition.

In a one million year time perspective, there is a small risk contribution from canister failures due to enhanced corrosion following buffer erosion

Loss of buffer may occur from exposure to low ionic strength waters but the extent is uncertain. The Forsmark site has a large potential to maintain a sufficient ionic strength at repository depth over a glacial cycle. Loss of buffer mass, to the extent that advective conditions arise in the deposition hole, may, however, occur in a 100,000 year perspective for typically less than ten deposition positions with high flow rates.

Advective conditions in a deposition hole will enhance the canister corrosion rate. In a one million year time perspective, this may lead to failures of a few canisters when applying the most pessimistic of the hydraulic interpretations made of the Forsmark site, with cautious assumptions regarding concentrations of corrosive agents and deposition hole acceptance rules.

With pessimistic assumptions regarding buffer erosion, copper corrosion and radionuclide transport conditions, the radiological risk from such canister failures is pessimistically calculated to be around 1/100 of the regulatory limit in a 100,000 year perspective and around 1/10 of the regulatory limit in a one million year time perspective.

In a one million year time perspective, there is a small risk contribution from canister failures due to earthquakes

Canister failures due to large earthquakes cannot be categorically ruled out. However, the probabilistic analyses imply that, on average, it would take considerably more than one million years for even one such canister failure to occur.

The contribution to radiological risk from earthquakes is pessimistically calculated to be less than 1/100 of the regulatory limit in a 100,000 year perspective and less than 1/10 of the regulatory limit in a one million year time perspective.

Issues related to altered climate conditions

Several issues of importance for long-term safety are related to future glacial, periglacial or warmer climate conditions. A number of conclusions regarding effects of such conditions can be drawn.

Freezing of the buffer is ruled out – even for very pessimistically chosen climate conditions

According to the analyses, freezing of the buffer clay is ruled out, even for the most pessimistic periglacial climate conditions considered, which includes the large uncertainties related to future climate development. Also freezing of a deposition tunnel backfill material or of a water-filled cavity in an eroded buffer is ruled out for the most pessimistic climate development at Forsmark.

Canister failure due to isostatic load is ruled out – even for very pessimistically chosen climate conditions

According to the analyses, canister failure due to isostatic load is ruled out for the most severe future glacial conditions considered based on the glacial development of the past two million years.

Oxygen penetration is very unlikely – even for very pessimistically chosen climate conditions – and the consequences are small

Oxygen penetration to canister positions is ruled out, except for enhanced flow situations occurring during the unlikely event when an ice sheet margin is temporarily stationary above the repository in combination with several other pessimistic assumptions. Even in such a case, the consequences in terms of canister corrosion are small.

Repository safety for a prolonged period of warm climate before the next glacial period is assessed as comparable to safety for a climate unperturbed by enhanced global warming

A prolonged period of warm climate (global warming due to an enhanced greenhouse effect) may lead to increased exposure to dilute groundwater of a repository at Forsmark and hence to increased buffer erosion. The sensitivity of the calculated risk to such a perturbation is, however, low.

The occurrence of large earthquakes is likely to increase during deglaciation, and this effect is thus delayed by a prolonged initial period of warm climate.

Other issues related to barrier performance and design

The reference design, forming the basis for the assessed initial state in SR-Site, yields a safe repository when implemented at the Forsmark site

Since the analyses in SR-Site show that the regulatory risk criterion is fulfilled, it is concluded that the assessed reference design implemented through the selected production and control procedures will yield a safe repository. Conclusions regarding design issues important for long-term safety yielding feedback to future refinement of the design have been drawn.

It is crucial to avoid deposition positions intersected by large or highly water conductive fractures and the low frequency of water conducting fractures allows efficient application of such rejection criteria

The risk contributors in SR-Site are related to the occurrence of large and/or highly conductive fractures intersecting deposition holes. This applies to the buffer colloid release process and the impact of major earthquakes in the vicinity of the repository. These two phenomena are related to canister failures due to canister corrosion and to secondary rock shear movements, respectively. As also the retention in large, highly transmissive fractures is small, such failures are in general associated with high consequences. Accordingly, such fractures need to be avoided once identified.

Cautious assumptions regarding the likelihood of occurrence of such fractures and regarding deposition hole rejection criteria are adopted in SR-Site. The results of the analysis are sensitive to these assumptions. It is important to continue the development of acceptance criteria for deposition holes as a basis for future assessments. This needs to be studied both by simulation of the effects of applying potential criteria and by exploring the practicability of applying the criteria.

The heat from the canister will likely fracture the rock in the deposition hole wall, which would enhance the in- and outward transport of dissolved substances, but this has little impact on risk

Thermally induced spalling around deposition holes at Forsmark cannot be ruled out and may have a considerable impact on mass exchange between the flowing groundwater and the buffer as long as diffusion is the dominant transport mechanism in the buffer. For diffusive conditions, there is, however, a considerable margin to canister failures even when spalling is pessimistically assumed in all deposition positions at Forsmark. If advective conditions prevail in the buffer, the effects of spalling are much less pronounced because it adds little to the already increased flow rate. In consequence, the overall effect on the calculated risk is small.

The importance of the excavation damaged zone in the rock around the deposition tunnels as a transport path for radionuclides is limited

The importance of the excavation damaged zone (EDZ) around deposition tunnels is limited in comparison to other transport routes for radionuclides. Very pessimistic assumptions about the EDZ in relation to the reference excavation method could affect the extent of canister corrosion for advective conditions.

This confirms the suitability of the cautious reference excavation methods adopted in the reference design of the repository.

In most deposition holes groundwater will not reach the canister for thousands of years due to the favourable rock properties at Forsmark

The saturation times for both backfill and buffer are likely to range from a few tens of years to several thousand years, as a consequence of the rock properties (matrix hydraulic conductivity and presence and characteristics of fractures) at Forsmark. The majority of deposition holes are not intersected by water conducing fractures, yielding slow saturation (with water from the deposition tunnel and the buffer) and slow inflow of e.g. corrosive agents in the groundwater both during unsaturated and saturated conditions. Transport of corrosive agents within the bentonite will be very limited and diffusion-dominated also during the unsaturated phase. During saturation the microbial activity may be enhanced before the swelling pressure is established. However, this is of little consequence, since the corrosion is limited by mass balance and transport. Since the groundwater flow during saturation is towards the deposition holes, no erosion of the bentonite can occur during the unsaturated period. The effects of the unsaturated period on the geochemical, mineralogical and thermal property changes in the buffer have been investigated in SR-Site. They are found to be small and without any significant impact on long-term performance.

S4.2 Demonstration of compliance

Introduction

This section summarises the most important aspects of the demonstration of compliance with applicable SSM regulations. A complete account of how the SR-Site report meets these requirements is given in Appendix A to the main report, where the regulations are reproduced and where references are given to sections in this report where each issue is addressed.

The safety concept and allocation of safety

As an introduction to the discussion of compliance, a brief account of the safety concept, evaluated by results from the SR-Site assessment is given. The main safety function of the KBS-3 concept is containment and the secondary safety function, mobilised if the containment function is not upheld, is retardation.

Containment for a KBS-3 repository at Forsmark

The containment function is provided by an intact copper shell of the canister. The extent to which this function is upheld is dependent on the buffer's function of limiting advective transport between the host rock and the canister and on favourable mechanical, hydrogeological and geochemical properties of the host rock: i) limited flow rates and a minimum charge concentration of the groundwater to avoid erosion of the buffer ii) limited flow rates and low groundwater concentrations of sulphide to limit corrosion, in particular if the buffer has been eroded, and iii) a low probability of large fractures intersecting deposition holes in order to limit the potential impact on the canister of large earthquakes in the vicinity of the repository.

The analyses in SR-Site indicate that containment is maintained even in the one million year perspective for a vast majority of canisters. Deterioration of the barrier system to the extent that containment is lost is assessed to only occur, as a statistical average, for less than one canister due to buffer erosion leading to advective conditions and enhanced corrosion. The other failure mode that could not be ruled out, that due to earthquake-induced secondary shear movements in fractures intersecting deposition holes, is even less likely and affects on average considerably less than one canister when this failure mode is evaluated statistically with a number of pessimistic assumptions. This means that containment is assessed to be maintained for the vast majority of the 6,000 canisters throughout the assessment period.

All safety functions related to containment are shown in Figure S-7. Many of these, like the canister's ability to withstand isostatic loads or the "ability" of the host rock to provide a favourable rock temperature, are assessed as upheld thought the assessment period.

It is also noted that the consequences of a postulated, complete loss of containment for all canisters decrease with time, and are about a factor of 3 higher than the regulatory risk limit at the end of the assessment period (hypothetical case C in Figure S-12).

Retardation for a KBS-3 repository at Forsmark

Both the failure mechanisms that could not be ruled out are of the common mode type, i.e. the canister, the buffer and the rock are all affected, either through a detrimental shear movement or through a high flow rate in the geosphere, affecting both erosion and corrosion. The causes of the failures affect also the retention properties through high flow rates and, in the case of erosion, through the absence of the buffer after failure. Hence the retarding potential of the repository is limited in these particular cases, for the canisters that have failed. Instead, safety is to a considerable extent achieved through the slow dissolution of the fuel and, to a lesser extent, through the limited corrosion rate of radionuclide-containing metallic structural parts of the fuel elements.

For the canisters that maintain their containment potential, retardation is a latent safety function throughout the assessment period. A more general view of the retarding potential of the buffer and the host rock is obtained from the analyses of hypothetical, complete losses of barrier functions above.

Retention in the buffer is important for the initial 1,000 years and limited in longer time frames (compare cases C and D in Figure S-12). The latter is due to the fact that the total dose in the long term is dominated by non-sorbing or very long lived nuclides. However, a nuclide specific comparison reveals that the buffer has a considerable retention function for sorbing nuclides also in the long term, masked in the total dose by the dominance of the non-sorbing species.

The role of retention in the rock is similar; it is important for the initial 1,000 years and limited in longer time frames (compare cases C and C* in Figure S-12) as concerns total dose, again since total dose is then dominated by nuclides that do not sorb in the rock. A nuclide specific comparison reveals that the rock has a considerable retention function for sorbing nuclides also in the long term, masked in the total dose by the dominance of the non-sorbing species. The low flow rates at repository depth also play an important role in limiting the release rate of radionuclides to the rock.

Summary

In summary, containment is the primary safety function of the KBS-3 repository and it is demonstrated to be efficiently upheld at the Forsmark site throughout the assessment period, directly through the properties of the canister and indirectly by the favourable hydrogeological and geochemical properties of the host rock. For the rare failures of containment, retardation is of limited importance due to the common mode nature of the failure mechanisms in question and since only very long-lived nuclides remain when these failures occur. As a latent function, retardation is significant for hypothetical releases of, in particular, sorbing species throughout the assessment period. For hypothetical failure modes affecting the canister only, retardation of sorbing nuclides is significant in the buffer and in the host rock.

Compliance with SSM's risk criterion

Compliance for the first 1,000 years

Of the three identified failure modes of the canister, i.e. failure due to corrosion, due to shear load and due to isostatic loads, corrosion and isostatic load induced failures can be ruled out with large margins for the first 1,000 years, as demonstrated in the reference evolution and the analyses of the corrosion and isostatic load scenarios.

Shear loads on the canister may occur as a consequence of large earthquake induced secondary shear movements in fractures intersecting deposition holes. Although the likelihood for large earthquakes is higher during periods of tectonic stress, for example during glacial rebound, it cannot be entirely ruled out that a detrimental earthquake would occur during the initial 1,000 years.

The probability that one out of the 6,000 canisters has failed at the end of the initial 1,000 year period is estimated at $2.4 \cdot 10^{-5}$, i.e. hypothetically 40,000 repositories, each with 6,000 canisters would have to be constructed in order for there to be an expectation of one failure during the initial 1,000 years. Despite this extremely low probability, a risk contribution was calculated for the first 1,000 years, with the result that the mean annual dose is at most around 0.001 μSv/yr corresponding to a risk of 10^{-10}/yr. This analysis builds on a detailed modelling of the biosphere development and radionuclide transport in the developing landscape during the initial 1,000 years, as required by SSM's Regulations.

It is, therefore, concluded that the analysed repository at Forsmark complies with the regulatory risk criterion during the initial 1,000 years after closure.

In hypothetical cases of initially defective canisters, there are several properties of the barrier system that provide protection, some of which are not important, or more difficult to claim, in longer time frames. Some of these would, with cautious assumptions, alone prevent any releases during the initial 1,000 years for an initially defective canister. This relates particularly to the time required for water to get into contact with the fuel elements and the integrity of the Zircaloy cladding.

Compliance for the time beyond the first 1,000 years

The bounding, dashed curve in Figure S-13 is the sum of the risk associated with the shear load scenario and that associated with the corrosion scenario. Each scenario is here pessimistically represented by the calculation case that yields the highest risk.

Other scenarios did not yield any contributions to the calculated risk. In the account of combinations of scenarios and phenomena it was concluded that the consequences of a shear load failure followed by buffer erosion needs to be assessed and this is done in the bounding case of the shear load scenario. All relevant risk contributions are, therefore, assessed to be included in Figure S-13.

Since the bounding curve in Figure S-13 is below the risk limit for the duration of the one million year assessment period, the analysed KBS-3 repository at the Forsmark site is assessed to fulfil the regulatory risk criterion. Risk dilution was shown not to challenge this conclusion.

It is furthermore concluded that a more realistic risk may be anywhere in the area below the bounding curve, down to zero risk based on the zero results of the three cases with no erosion and a situation where no canisters would fail due to shear movements induced by large earthquakes, which could be reached if somewhat less pessimistic assumptions could be defended for the shear load scenario.

It is also noted that through the use of LDF factors to transform releases to doses, it is in the risk assessment implicitly assumed that the landscape to which the releases occur is always fully populated, including the object where the highest dose is calculated to occur and during the point in time when this occurs. Furthermore, long periods of glacial and submerged conditions are expected where no doses to humans occur since the site is not habitable. This has not been taken into account in the risk summation. Rather, temperate conditions yielding the highest doses are assumed.

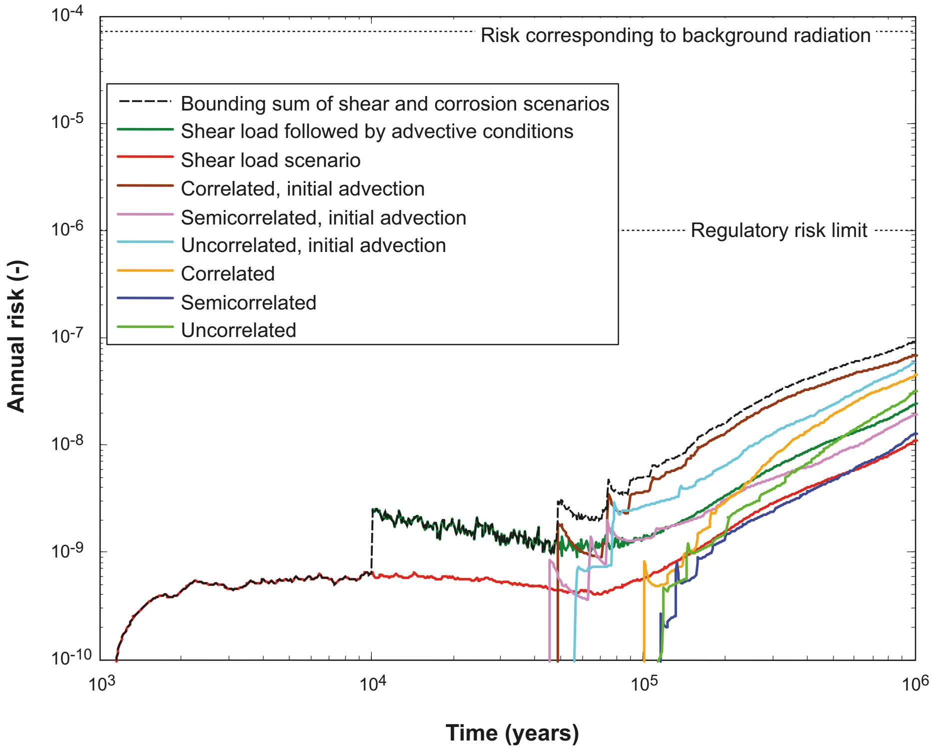


***Figure S-13.** Risk curves, expressed as annual individual risk. Several alternatives for the corrosion scenario are shown, and two for the shear load scenario. The bounding, dashed curve is the sum of the curve for the shear load failure followed by advective conditions (dark green) and the curve for the variant of the corrosion scenario yielding the highest risk (brown). The risk associated with the main scenario is subsumed under the corrosion scenario as it is equal to the semi-correlated case (blue).*

The time beyond one million years

Although the rocks, the ductile deformation and the brittle deformation zones at Forsmark all formed at least one billion years ago, the increasing uncertainties regarding the external conditions makes it not meaningful to predict the development of the site and of the repository for time periods beyond one million years.

It is noted that the hypothetical case in Section S3.11, where all canisters are failed, where the buffer is absent and where retention in the rock is disregarded, yields releases that correspond to dose consequences that are comparable to the natural background radiation after one million years. The releases are in that case controlled by the flow at repository depth and the inventory of radionuclides, the latter of which decreases with time and is dominated by radionuclides that also occur in natural ore bodies.

Finally, the scientific understanding of the fuel dissolution process suggests that the longevity of the spent fuel matrix is several million years in the repository environment. Furthermore, uranium ores that are many millions and even billions of years old are known through geological observations. This indicates that, in the case of stable tectonics and maintained reducing conditions in the repository, the uranium oxide which constitutes the fuel matrix can be stable for many millions of years.

Alternative safety indicators

In particular for times far into the future, the calculated risk becomes less useful as an indicator of repository safety, and SSM's Regulations suggest that alternative indicators should, therefore, also be evaluated. Four alternative indicators to risk are used in SR-Site. The following results emerged for the central corrosion case.

- Peak releases of activity from the geosphere are about three orders magnitude below the activity constraints issued by the Finnish regulator STUK.
- The peak radiotoxicity flux from the geosphere is more than three orders of magnitude lower than the reference value for the radiotoxicity flux from the geosphere suggested by the EU SPIN project.

- Calculated radionuclide peak concentrations in ecosystems at Forsmark from repository releases of Ra-226 are about three orders of magnitude below measured concentrations of naturally occurring Ra-226 at Forsmark.
- Peak geosphere fluxes caused by Ra-226 releases from the repository are about two orders of magnitude below naturally occurring fluxes of Ra-226 at the site, as estimated from site data; the difference is larger for U-234 and U-238, see Figure S-14. The total release of all repository derived nuclides converted to dose is also around two orders of magnitude lower than the summed dose from fluxes of the three mentioned naturally occurring nuclides.

The results are readily applicable to other corrosion cases, for most indicators by simply scaling with the release of Ra-226, which is at most one order of magnitude higher than that for the central corrosion case. Similar conclusions are reached for the shear load scenario.

In summary, the application of alternative indicators shows that releases from the repository are orders of magnitude below the adopted reference values for the indicators. This suggests that the future radiological consequences on man and on the environment of releases from the repository are negligible, independent of assumptions in the biosphere model.

Uncertainties linked to the risk calculation for different time periods

Uncertainties that have a significant influence on the calculated risk have been identified for the corrosion and the shear load scenarios. The handling of the uncertain factors in SR-Site is summarised together with references to plans for the reduction of these uncertainties. Most of the uncertain factors have been treated pessimistically, whereas some have been included as probability distributions in the risk calculations, where their full uncertainty range is used in the determination of mean annual doses, the relevant metric for determining the calculated risk.

The combination of pessimistic handling of uncertainties for which probability distributions could not be determined with the probabilistic handling of quantified uncertainties means that the total risk as determined in the risk summation is claimed to represent an upper bound on risk. Since this upper bound is below the risk limit throughout the one million year assessment period, there are no uncertainties of critical importance to resolve with respect to risk.

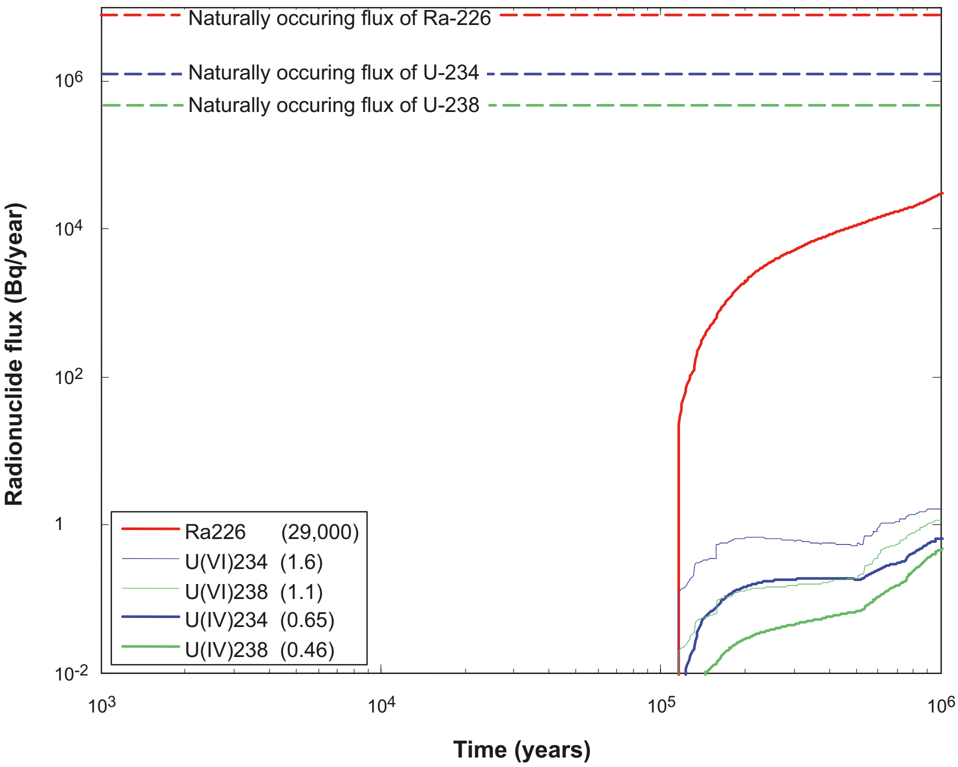


Figure S-14. *Far-field release rates (Bq/year) of U-238, U-234 and Ra-226 in the central corrosion case compared to the naturally occurring fluxes at Forsmark.*

Effects on the environment from release of radionuclides

Doses to non-human biota for the central corrosion case have been calculated. The highest dose rates to organisms in marine, freshwater and terrestrial ecosystems in Forsmark, both in total and for the dose-dominating radionuclides are well below the screening dose rate (10 µGy/h) recommended in the ERICA Integrated Approach. It is therefore concluded that radionuclide releases predicted for this case will not lead to significant detrimental biological effects on individuals of species found, or projected to occur in future, at the site. In consequence, there will be no detrimental impact on populations, communities or ecosystems. The conclusion is applicable to all other cases included in the risk summation.

The use of best available technique, BAT

According to Swedish legislation, a licence application for a final repository needs to address the issue of best available technique (BAT). While a general account of the use of BAT is a broad issue spanning from the selection of method for the management of nuclear waste to fine details of the selected method, a limited part of this issue can and should be addressed in the safety assessment of the preferred method. Feedback can be given whether alterations in relation to the analysed reference design could lead to reductions in risk or in reduction in uncertainties that potentially could affect risk. For aspects of the design where no such reduction in risk or uncertainty in fulfilment of safety functions can be seen to be realistically obtainable, SR-Site claims the solution to be optimal and BAT. However, SR-Site is not an assessment of all conceivable technical solutions. SKB will continue technical development of several aspects of the design in order to further simplify construction and implementation, but will only adopt these developments if they lead to a risk comparable to or lower than that found in SR-Site. The following is concluded regarding BAT.

- There seems to be little need to alter the *canister design* from a safety perspective. However, the ability of the cast iron insert to withstand shear load depends mainly on the occurrence of surface defects in the insert. While it is concluded in the **Canister production report** that the current canister reference design conforms to the stated design premises, it is also noted that rigorous requirements on manufacturing and NDT (non-destructive testing) capability are needed. Furthermore, the adequacy of inserts for PWR fuel have not formally been assessed, but the PWR design is more robust due to its larger material thickness in the cast iron insert.
- There is potential to further reduce risk due to shearing by further reductions of the maximum allowed buffer density and adapting the production of the buffer and deposition holes to conform to this design. While there still is uncertainty with regard to modelling of colloid formation and subsequent erosion of the buffer material and the modelling approach thus tends to be pessimistic, it cannot with the current understanding, be defensibly mitigated by e.g. selection of another buffer material. A continued R&D programme on buffer erosion mechanisms is needed.
- There possibly could be improvements in the backfill design from an installation point of view, but there does not seem to be a need to change the design to further improve its safety functions.
- The application of deposition acceptance criteria in the form of EFPC provides adequate protection against shear failure, but there is reason to continue the efforts envisaged in the detailed investigation programme since the EFPC unnecessarily implies rejection of many holes only intersected by short fractures. Finding other means of identifying large fractures could increase the efficiency in the criterion without loss of safety and also increase the safety level since the focus on the detailed investigations would be on the few fractures of potential concern. Application of EFPC is also important for protection against corrosion failures, but there appears to be a potential for further enhancing safety by avoiding deposition holes with high inflows.
- The selected repository depth is adequate and changing the depth is not deemed to reduce the calculated risk. Furthermore, a shallower depth, e.g. above the 400 m level might increase the risk, since the frequency of water conducting fractures is higher there. Placing the repository some 100 m deeper would probably result in a risk contribution similar to the one obtained from the selected depth, whereas much deeper locations would imply that additional factors, such as very high stress levels, might need to be considered.

S4.3 Feedback from the analyses in SR-Site

Design premises

A sufficient level of safety for a repository at Forsmark constructed in accordance with the reference design and conforming to the design premises is demonstrated in SR-Site. Thus, most of the already stated design premises and associated reference designs are judged adequate as they are, whereas some are worthy of reconsideration. The most important suggested revisions of the design premises are the following.

- Current design premises on isostatic load should be revised to consider the uncertainty in future ice thickness.
- The findings from SR-Site suggest that the risk contribution from shear loads on the canister would be further reduced if the buffer density was lower than the maximum value of 2,050 kg/m^3. Experience from the buffer design work suggest that it would be possible to replace the upper limit with a distribution of allowed densities such that only a small fraction of deposition holes would have densities close to 2,050 kg/m^3. Such a distribution could also consider the montmorillonite content of the dry buffer material.
- Given the issues relating to the current design of the bottom plate, alternative solutions should be investigated. Ideally, only the canister and bentonite should be allowed in a deposition hole.
- According to current design premises deposition holes should, as far as reasonably possible, be selected such that they do not have potential for shear larger than the canister can withstand. It is noted that the EFPC criterion is a tool to identify such large fractures, but can be replaced or complemented by other tools if these are shown to yield a comparable or lower risk.
- The analyses in SR-Site suggest that the calculated risk would be reduced if it were possible to identify the deposition holes having the largest Darcy fluxes during saturated conditions. It is suggested that the current design premises could be revised such that deposition holes with potential for high Darcy flux during the assessment period are avoided. The practical formulation of the rule needs further elaboration. A tentative rule may be to avoid deposition holes intersected by connected transmissive fractures capable of producing higher inflow than 0.1 L/min. This also means that that deposition holes intersected by fractures showing visible grout should also be rejected. The criterion should be combined with the application of the EFPC, for the entire deposition hole, but only for fractures showing potential for groundwater flow.
- The variation cases analysed in SR-Site confirm that the suggested upper limit of the connected EDZ transmissivity of 10^{-8} m^2/s in deposition tunnels is adequate and need not be decreased. However, connected EDZ transmissivity above this value will start to affect risk and needs to be avoided.

Detailed site investigations

While the confidence in the site understanding is judged adequate and remaining uncertainties are sufficiently constrained to allow bounding risk estimates, the repository might still be further optimised with respect to efficiency and risk reduction. Many of such potential improvements concern possibilities for local adaptation of the deposition tunnels and deposition holes to the conditions found in the rock, although some issues remain concerning the properties of the rock mass outside the immediate vicinity of the deposition areas. Feedback to the detailed site investigations and site modelling can thus be given in order to prioritise and further specifying the development needs in existing plans.

Plans for the detailed investigation programme have been developed by SKB. Within these plans, the results in SR-Site highlight the needs for:

- Determining the actual extent of the damage zone for the few deformation zones able to host larger earthquakes at the Forsmark site.
- Developing means for bounding the size of fractures intersecting deposition holes.
- Reducing the uncertainty in the geological DFN model.
- Developing means of finding (and then avoiding) connected transmissive fractures capable of producing higher inflows and high Darcy fluxes into and around deposition holes.
- Reducing the uncertainties in the actual distribution of hydraulic properties within water conductive fractures.

- Developing the method to control the EDZ as well as demonstrating the reliability of this method.
- Characterising both the rock stress and the spalling strength of the rock.
- Enhancing the confidence in evaluations of future sulphide levels.
- Supplementing data for some parameters for the biosphere modelling.

Generally, the plans presented in the detailed investigation framework programme are judged adequate in relation to these suggestions.

Research on processes of relevance for long-term safety

In accordance with a proper safety culture, general research on processes of importance for safety should continue, even if current knowledge is sufficient to demonstrate long-term safety. More specifically, there are some issues which SR-Site shows to contribute to risk and where the basis for the assessment can be improved through more R&D.

- For the spent fuel, such issues include continued efforts to understand spent fuel dissolution mechanisms and to quantify i) the gap and grain boundary inventory and ii) the corrosion release rates from the metal parts of the waste.
- Understanding copper corrosion is fundamental for the safety concept. While confidence is judged high, it is still essential to continue ongoing research. For the canister understanding the deformation of cast iron is essential for confidence in the assessment of the impact of mechanical load on the canister. Understanding of copper deformation under external pressure is also essential for confidence in the assessment of the impact of mechanical load on the canister. Conditions for stress corrosion cracking are not judged to occur in the repository, but the basic research focusing on identifying the necessary conditions needed for stress corrosion cracking to occur on copper should continue.
- Regarding the buffer, additional research on water transport, gas transport, piping/erosion, homogenisation and self healing, montmorillonite alteration and the studies on rheological effects of cementation should be continued. From the studies of buffer erosion/colloid release it has been concluded that the process cannot easily be excluded and that a continued R&D programme is needed. Further studies could reduce the degree of pessimism currently adopted.
- Much of the R&D related to the geosphere is covered by the detailed investigation programme, but research on some processes should continue. Research areas of interest include efforts to handle and mitigate thermally induced spalling, continued research on coupled thermo-hydro-mechanical processes in rock, assessing the potential for earthquakes including further development of the earthquake simulation tools, developing the DFN methodology, better bounding of the expected evolution of sulphide at the Forsmark site, the role of microbial activity for maintaining a low and stable redox potential, for sulphide formation and oxygen consumption, and the role of channelling for radionuclide migration.

There are also a number of issues regarding the biosphere and future climate of interest to take forward in the RD&D Programme.

S4.4 Confidence in the results of the assessment

The SR-Site assessment supports SKB's application for a final repository, i.e. a major decision point in SKB's programme for the management of spent nuclear fuel. A statement on the confidence in the results obtained in SR-Site is, therefore, appropriate. The confidence in the results obtained is assessed as sufficient for the decision at hand based on the following.

- The knowledge of the Forsmark site from the completed, surface based investigations is sufficient for the assessment of long-term safety. The site has favourable conditions for safety and no site-related issues requiring resolution in order to demonstrate safety have been identified. Confidence in the site-descriptive model and in the understanding of the site is obtained by a systematic and quality assured programme for site investigations and site modelling. The confidence in the site model is assessed in detail and documented in the **Site description Forsmark** report.

- There is a well established reference design with specified and achievable production and control procedures yielding an initial state of the repository system with properties favourable for long-term safety at the Forsmark site. The engineered parts of the repository system are based on demonstrated technology and established quality assurance procedures to achieve the initial state of the system. This is systematically documented in the **Production reports** and their underlying references. Examples of important aspects of the initial state of the engineered barriers include:
  a. The copper canister sealing quality.
  b. The cast iron insert casting quality.
  c. Buffer properties such as density and content of montmorillonite and impurities.
  d. Backfill properties ensuring its ability to keep the buffer in place and to swell.
  e. The quality of the approach to adapt the repository to the detailed conditions found underground and the quality of the excavation technique.
  f. The quality of the deposition technique.

There is potential for additional optimisation when this reference design is developed and implemented.

- The scientific understanding of issues relevant for long-term safety is mature as a result of decades of research both within the Swedish and other national programmes and in international collaboration projects. The R&D efforts to understand repository evolution and safety have led to the understanding of key processes like copper corrosion, shearing of canisters and other potential canister failure causes, and of key phenomena controlling retardation. This knowledge is, in SR-Site, systematically documented in several reports in a format suitable for use in the safety assessment.

- The SR-Site main report and its supporting documents have undergone comprehensive peer reviewing. In particular the scientific basis of the safety assessment has undergone review by recognised experts in the relevant fields of science.

- A complete analysis of issues identified as relevant to long-term safety has been carried out in SR-Site according to an established assessment methodology, comprising e.g. cautious approaches when addressing uncertainties.
  – The understanding of safety is built on a systematic identification of safety functions and criteria for the safety functions.
  – Repository evolution is analysed with a structured approach in several time frames, addressing in each of these the processes that have been identified as relevant and with the safety of the system, as expressed by the safety functions, as a focus. Data uncertainties and data quality are assessed and documented according to a pre-established template. Quality assurance of models and modelling is achieved by following procedures documented in a **Model summary report**. The assessment is then broken down into a set of scenarios to exhaustively scrutinise all possible ways in which the identified safety functions could be impaired and consequences of such situations. Additional arguments and analyses are provided according to Section S3.12.
  – Confidence in the key results of radionuclide transport and risk calculations is enhanced by the fact that they can often be closely reproduced with simple, analytical models, using the same input data as the fully qualified numerical models.
  – The key results of radionuclide transport and risk calculations are overestimates since a number of pessimistic assumptions were made in the analyses, both regarding the extent of canister failures and regarding their consequences.

- Documented quality assurance routines have been applied in the assessment of the initial state, in the development of the site description and in the analysis of long-term safety. A QA-plan, encompassing most of the routines followed in undertaking the steps described in the above points, has been established and implemented in SR-Site. This is part of the overall methodology followed in the assessment.

S5 Overview of the main report

Following the introductory Chapter 1, this report outlines the methodology for the SR-Site assessment in Chapter 2, and presents in Chapter 3 the handling of features, events and processes, FEPs, of importance for long-term safety. Chapter 4 presents the site and Chapter 5 the initial state of the constructed repository. Chapters 6 and 7 present the plans and methods for handling external influences and internal processes, respectively. Safety functions and safety function indicators are discussed in Chapter 8. The collection of input data for the assessment is described in Chapter 9. The material presented in the first nine chapters is utilised in the analysis of the reference evolution in Chapter 10, focussing on the containment potential of the repository. Scenarios for the further evaluation of safety are selected in Chapter 11. The selected scenarios are analysed in Chapter 12 with respect to containment potential and in Chapter 13 with respect to retardation potential, through radionuclide transport and dose assessments. Additional analyses supporting the safety assessment are presented in Chapter 14. Conclusions and feedback are provided in Chapter 15.

A list of references is given in Chapter 16. Appendix A is an account of how applicable regulations are addressed in the assessment. A glossary of abbreviations and specialised terms used in SR-Site is found in Appendix B and topography and place names in the Forsmark area are provided in Appendix C.

1 Introduction

1.1 SKB's programme for spent nuclear fuel

Radioactive waste from nuclear power plants in Sweden is managed by the Swedish Nuclear Fuel and Waste Management Co., SKB. Within SKB's programme for the management of spent nuclear fuel, an interim storage facility and a transportation system are today (March 2011) in operation. Several decades of research and development has led SKB to put forward the KBS-3 method for the final stage of spent nuclear fuel management. In this method, copper canisters with a cast iron insert containing spent nuclear fuel are surrounded by bentonite clay and deposited at approximately 500 m depth in groundwater saturated, granitic rock, see Figure 1-1. The purpose of the KBS-3 repository is to isolate the nuclear waste from man and the environment for very long times, see further Section 2.4. Around 12,000 tonnes of spent nuclear fuel is forecasted to arise from the currently approved Swedish nuclear power programme (where the last of the 10 operating reactors is planned to end operation in 2045), corresponding to roughly 6,000 canisters in a KBS-3 repository.

Two principal remaining tasks in the programme are to build and operate i) the final repository and ii) an encapsulation plant in which the spent fuel will be emplaced in canisters to be deposited in the final repository.

SKB has carried out site investigations for a final repository in the municipalities of Östhammar (Forsmark area) and Oskarshamn (Laxemar area), Figure 1-2. In June 2009, the Forsmark site was selected by SKB as the site for the final repository. This report is the main report of the assessment of long-term safety for the licence application to build a final repository at the Forsmark site. The selection of the Forsmark site is justified in a separate report supporting the licence application /SKB 2010e/.

SKB has also selected Oskarshamn for the location of the encapsulation plant, in conjunction with the existing interim storage facility. An application for the encapsulation plant was made in November 2006 and amended in October 2009.

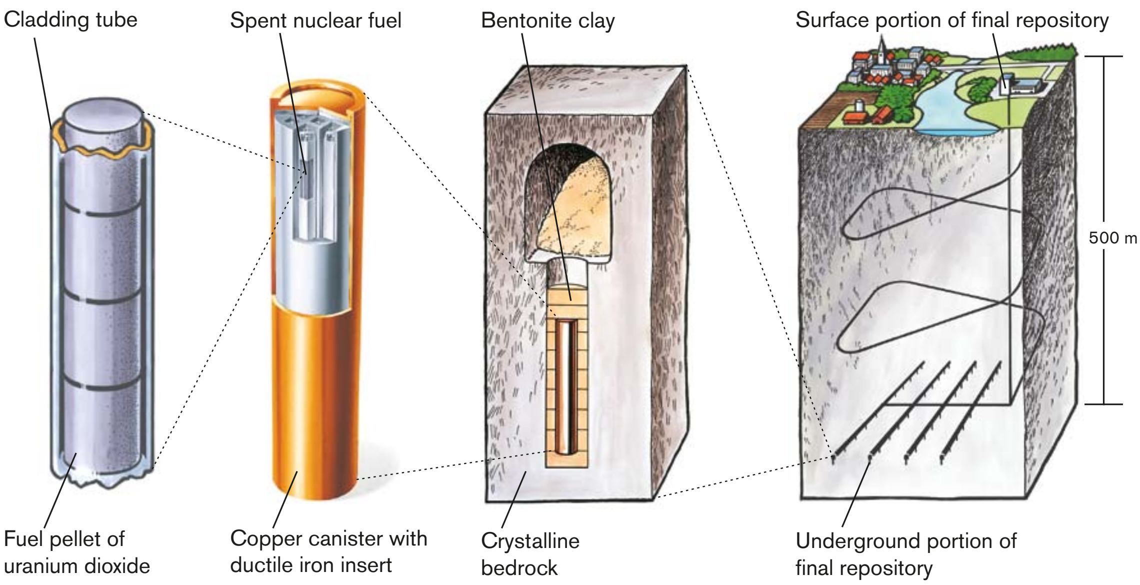


Figure 1-1. *The KBS-3 concept for disposal of spent nuclear fuel.*

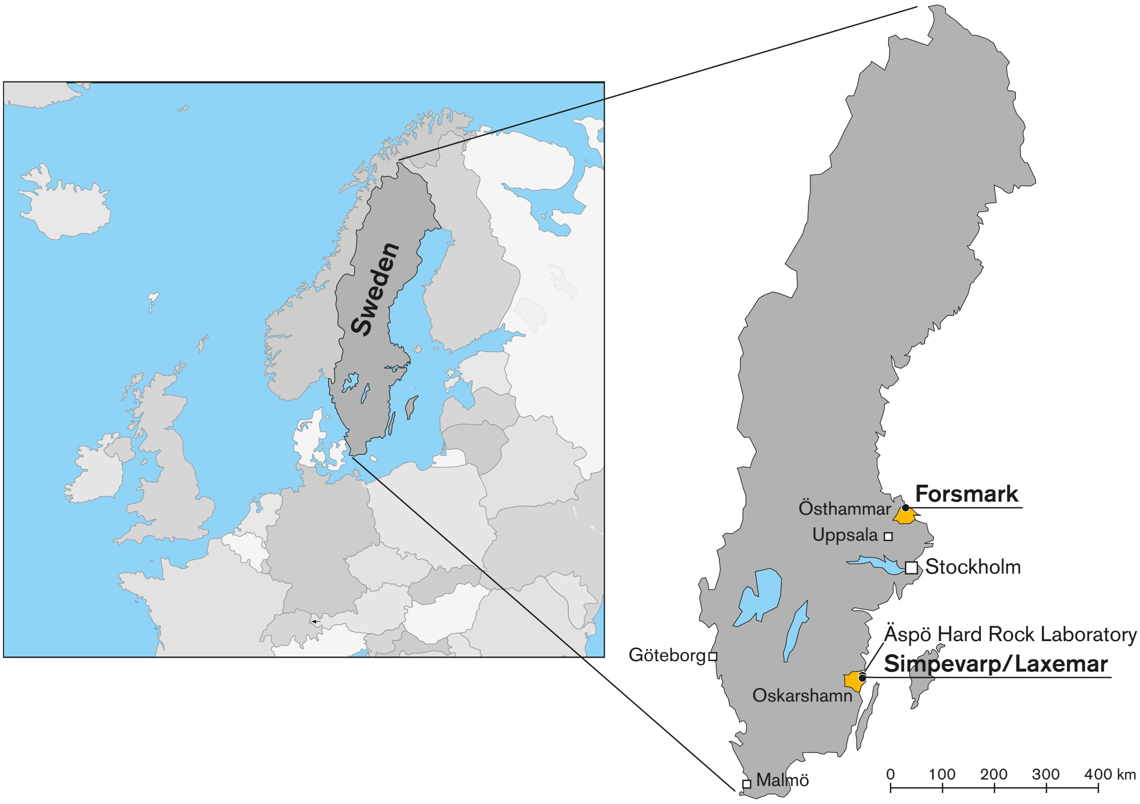


Figure 1-2. *The locations of the two investigated sites for a final repository, Forsmark in the municipality of Östhammar and Laxemar in the municipality of Oskarshamn.*

1.1.1 The role of the SR-Site report in the licence application

This SR-Site report constitutes a part of SKB's licence applications to construct and operate a final repository for spent nuclear fuel at Forsmark. SKB will submit two licence applications, one according to the Act on Nuclear activities and one according to the Environmental Code, and the SR-Site report is included in both applications, see Figure 1-3.

The licence documentation consists of an application document and a set of supporting documents. Important supporting documents are the Environmental Impact Assessment and the Safety Report. The Safety Report consists of a top document summarising and integrating various aspects of radiological safety during operation and after closure of the repository facility.

The safety report has two sub-reports that provide more detailed assessments of safety. These are the safety report for the operational period, SR-Operation, and the present SR-Site report, which deals with the long term safety after closure. The SR-Operation report describes how safety is maintained during operation and closure and also the measures taken to ensure that operation provides the initial state analysed in this report while the SR-Site report demonstrates safety in the long term. The SR-Operation report includes an analysis of potential mishaps that, if undetected, could impact on long-term safety, and no such risk of undetected mishaps was identified. A detailed account of production and operational procedures and quality control of importance for long term safety is given in several **Production reports** which are joint references to the SR-Site and SR-Operation reports. The SR-Operation report is thereby not required as a reference to SR-Site and is therefore not further cited in this report.

The licence applications also include special supporting documents on the selection of the KBS-3 method for final disposal of spent nuclear fuel and the selection of the Forsmark site. The KBS-3 method has two variants, vertical and horizontal emplacement of the waste canister. SKB has together with Posiva, the Finnish waste management organisation, studied horizontal emplacement. Horizontal emplacement for KBS-3 appears feasible but essential technology development and research remains in order to resolve critical issues to the same extent as has been done for KBS-3V. The SR-Site project and the licence application relate to the vertical emplacement mode shown in Figure 1-1.

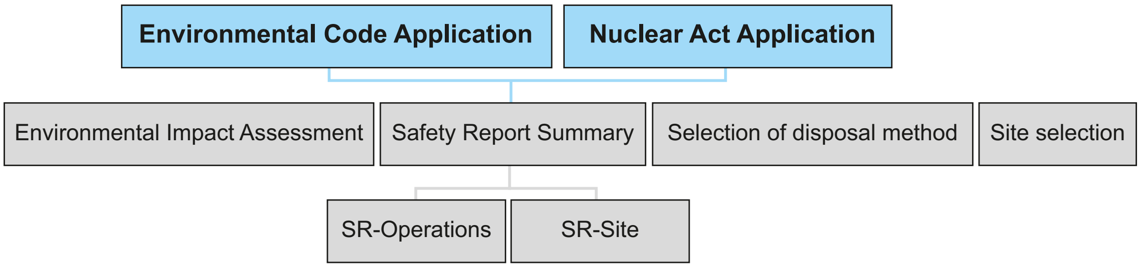


Figure 1-3. *Simplified structure of SKB's licence applications to construct and operate a final repository for spent nuclear fuel at Forsmark.*

1.2 Purpose of the SR-Site safety assessment project

As mentioned in Section 1.1.1, the role of the safety report SR-Site in SKB's licence application is to demonstrate that the repository SKB intends to build at the Forsmark site is safe in the long term. The purpose of the safety assessment project SR-Site is to investigate whether a safe repository can be built at Forsmark. If safety is demonstrated, then the SR-Site report can serve its intended purpose in the licence application.

The main purposes of the safety assessment project SR-Site are:

- To assess the safety, as defined in applicable Swedish regulations, of the proposed KBS-3V repository at Forsmark.
- To provide feedback to design development, to SKB's RD&D Programme, to detailed site investigations and to future safety assessment projects.

While SKB has established a technically feasible reference design and layout of the KBS-3V repository and showed that this conforms to stated design premises, see further Chapter 5, technical development will continue. Detailed designs adapted to an industrialised process designed to fulfilling specific requirements on quality, cost and efficiency need still be developed. The layout needs to be adapted to the local conditions found when constructing the repository at depth. These, potentially more optimal solutions, should result in at least the same level of safety as the current reference design being assessed in SR-Site.

Safety assessments at various stages of the programme will draw on the information available at that particular stage. Information on all the system components is needed at every stage, since safety depends on all these elements. The focus of a particular assessment, however, will be determined not only by the information available but also by the purpose of the assessment, i.e. the decision or decisions that it is intended to support.

The SR-Site report supports the licence application. The objective of the SR-Site report is to investigate whether the KBS-3 method has the potential of fulfilling regulatory safety criteria at the Forsmark site, with the host rock conditions determined from the surface based site investigations. The assessment is based on a reference design of the engineered parts of the repository, including reference methods to achieve the specified design, taking into account methods of controlling that the specifications of the reference design have been achieved. Another important purpose is to give feedback for further developments to that specification, in particular by giving input to the development of updated design premises.

1.3 Feedback from the SR-Can report

An important step leading up to the present report was the preparation of the SR-Can safety assessment report, published in November 2006 /SKB 2006a/. The main purposes of that report were to obtain a first assessment of long-term safety of a KBS-3 repository with vertical emplacement at the Forsmark and Laxemar sites, based on data from an initial site investigation stage, and to foster a dialogue with responsible authorities regarding interpretations of the applicable regulations and the acceptability of the approach adopted for safety assessment. The SR-Can report was reviewed by the Swedish Nuclear Power Inspectorate (SKI) and the Swedish Radiation Protection Authority (SSI) in 2007 and 2008[2].

[2] SKI and SSI were merged into the Swedish Radiation Safety Authority (SSM) 1 July 2008.

The present SR-Site report builds on the SR-Can assessment in several important respects:

1. The methodology and structure of the assessment and its reporting is based on SR-Can.
2. The outcome of the review of the SR-Can assessment has been taken into account when updating the methodology and scope of the assessment, see further Section 1.3.1.
3. The concluding Chapter 13 of the SR-Can main report pointed to a number of needs for development of process understanding, quantitative models and data, and issues related to safety assessment methodology. Addressing these has led to results incorporated in the present report.
4. The results of the SR-Can analyses were used in subsequent sensitivity analyses /Hedin 2008a/, to formally identify processes and data of importance for the risks calculated in SR-Can. The findings are in agreement with those in the SR-Can main report and thus support the identification of important factors for SR-Site derived in the SR-Can report.
5. The results of the SR-Can assessment have been used to give feedback to the design development of the KBS-3 concept. The feedback given in the concluding Chapter 13 of the SR-Can report has been extended /SKB 2009a/ and is to some extent based on complementary analyses carried out after the SR-Can assessment. In particular, more detailed design premises have been formulated for the canister, the buffer and the deposition tunnel and on how to adapt the design and layout to the host rock. These requirements were considered when establishing the repository design analysed in SR-Site, see further Chapter 5.

1.3.1 Review

The SR-Can report has been reviewed by SKI and SSI /Dverstorp and Strömberg 2008/, aided by evaluations by three international review teams focussing on safety assessment methodology /Sagar et al. 2008/, the evaluation of the engineered barrier system /Savage et al. 2008/ and the use of site investigation data /INSITE/OVERSITE 2008/, respectively. The authorities also employed a number of experts to assess more specific issues not covered by the other review teams. The main conclusions from the authorities' summarising review report are the following /Dverstorp and Strömberg 2008/:

- SKB's methodology for safety assessment mainly complies with the authorities' regulatory requirements, although parts of the methodology need to be further developed prior to a licence application.
- SKB's quality assurance of the safety assessment is insufficient in SR-Can.
- Prior to the licence application, a better knowledge base is needed with respect to certain critical processes with a potentially great impact on the risk from the repository, including erosion of the buffer in deposition holes.
- SKB needs to confirm that the assumed initial state of the repository is realistic and achievable.
- Reporting of the risk of early releases should be strengthened.

All conclusions from the review have been considered in detail in the preparation of this report. This is in accordance with the QA procedures for SR-Site. These require that findings in regulatory reviews of previous assessments are considered. The review findings have been used to identify a large number of items that are addressed in a structured way in the SR-Site assessment. The documentation of these items and their handling in SR-Site forms part of the project documentation and is made available to reviewing authorities on request, but is not issued as an SKB report.

1.4 Regulations

The form and content of a safety assessment, and above all the criteria for judging the safety of the repository, are defined in regulations originally issued by SKI and SSI and now by SSM. The regulations are based on various pertinent components of framework legislation, the most important being the Nuclear Activities Act and the Radiation Protection Act. Guidance on radiation protection matters is provided by a number of international bodies, and national legislation is often, as in the case of Sweden, influenced by international recommendations.

Regarding long-term safety of nuclear waste repositories, there are two more detailed regulations of particular relevance, issued under the Nuclear Activities Act and the Radiation Protection Act, respectively:

- "The Swedish Radiation Safety Authority's Regulations concerning the Protection of Human Health and the Environment in connection with the Final Management of Spent Nuclear Fuel or Nuclear Waste" (SSMFS 2008:37).
- "The Swedish Radiation Safety Authority's regulations concerning safety in final disposal of nuclear waste" (SSMFS 2008:21).

These two documents are reproduced in their entirety in Appendix A to this report. The way in which this SR-Site report addresses the requirements is indicated by references to relevant sections of this report, as inserts in the regulatory texts in the Appendix.

Potential risks to human health and the environment due to chemically toxic materials in the repository are addressed in the Environmental Impact Assessment.

1.4.1 Regulations for final disposal of spent nuclear fuel, SSMFS 2008:37

The parts of SSMFS 2008:37 most relevant to an assessment of long-term safety imply the following:

- Protection of human health shall be demonstrated by compliance with a risk criterion that states that "the annual risk of harmful effects after closure does not exceed 10^{-6} for a representative individual in the group exposed to the greatest risk". "Harmful effects" refer to cancer and hereditary effects. The risk limit corresponds, according to SSM, to an effective dose limit of about $1.4 \cdot 10^{-5}$ Sv/yr. This, in turn, corresponds to around one percent of the natural background radiation in Sweden.[3]
- Regarding environmental protection, biological effects of ionising radiation due to releases of radioactive materials from the repository in living environments and ecosystems of relevance shall be described, based on available knowledge.
- The consequences of intrusion into a repository shall be reported and the protective capability of the repository after intrusion shall be described.
- SSM requires a more detailed assessment for the first 1,000 years following repository closure than for later times.

SSM has also issued General Guidance concerning the application of SSMFS 2008:37. There, more detailed information regarding the above aspects is given. The General Guidance is also reproduced in Appendix A.

In the General Guidance, it is indicated that the time scale of a safety assessment for a final repository for spent nuclear fuel should be one million years after closure, see further Section 2.4. A detailed risk analysis is required for the first thousand years after closure. Also, for the period up to approximately one hundred thousand years, the reporting is required to be based on a quantitative risk analysis.

For the period beyond one hundred thousand years, the General Guidance to SSM 2008:37 states that a strict quantitative comparison of calculated risk in relation to the criterion for individual risk in the regulations is not meaningful. Rather, it should be demonstrated that releases from both engineered and geological barriers are limited and delayed as far as reasonably possible using calculated risk as one of several indicators. The demonstration of this is in the regulation seen as part of the demonstration of the use of best available technique, BAT.

[3] The natural background radiation in Sweden is around 1 mSv/yr /SSI 2007/.

1.4.2 Regulations concerning safety in final disposal of nuclear waste, SSMFS 2008:21

The parts of SSMFS 2008:21 most relevant to an assessment of long-term safety imply the following requirements.

- A safety assessment shall take into account features, events and processes (FEPs) that can lead to the dispersion of radioactive substances after closure.
- A safety assessment shall cover as long a period as barrier functions are required, but at least ten thousand years.
- Reporting of
  - analysis methods for system description and evolution,
  - analysis methods for the selection of scenarios (including a main scenario that takes into account the most probable changes in the repository and its environment),
  - the applicability of models, parameter values and other conditions used in the analyses,
  - handling of uncertainties and sensitivity analyses.
- Regarding analysis of post-closure conditions, SSM requires descriptions of the evolution of the biosphere, geosphere and repository for selected scenarios; and evaluation of the environmental impact of the repository for selected scenarios, including the main scenario, with respect to defects in engineered barriers and other identified uncertainties.

SSM has also issued General Recommendations concerning the application of SSMFS 2008:21. There, more detailed information regarding e.g. classification of scenarios and uncertainties is given. Excerpts from the Recommendations, relevant to an assessment of long-term safety, are also given in Appendix A, along with a statement of how this SR-Site report addresses the requirements.

1.5 Organisation of the SR-Site project

The SR-Site project started in April 2007, after the completion of the SR-Can project in December 2006. Throughout the SR-Site project, a number of specialists at SKB as well as external consultants have formed a core group of the project. The group consists of several generalists in the field of safety assessments of nuclear waste repositories and a number of experts in key areas of importance for the assessment. The individuals presently forming the SR-Site team and their roles are listed in the preface to this report. A large number of external experts have also contributed, mainly by producing specialised analyses and documentation of the scientific basis for the assessment. The roles, experience etc. of both the SR-Site team and of the external experts are documented, see further Section 2.9.4.

1.6 Related projects

The safety assessment project is closely linked to site investigation and engineering activities at SKB, see Figure 1-4.

1.6.1 Site investigations and site modelling

A considerable part of the basis for the safety assessments SR-Can and SR-Site is provided from SKB's completed site investigations in the municipalities of Oskarshamn and Östhammar.

Field data from the site investigations were analysed, within the site investigation project, by a site analysis team that produced a sequence of site descriptive models of the geosphere and the biosphere. The team consisted of several groups specialising in different disciplines. The site descriptive model is a synthesis of observations of the current state of the site and of the understanding of past and ongoing e.g. hydraulic and geochemical, processes driven by phenomena such as land uplift and climate change. Model simulations of the historical evolution of the site are an important part of the synthesis work carried out by the site analysis group. The resulting geosphere 3D model of current conditions

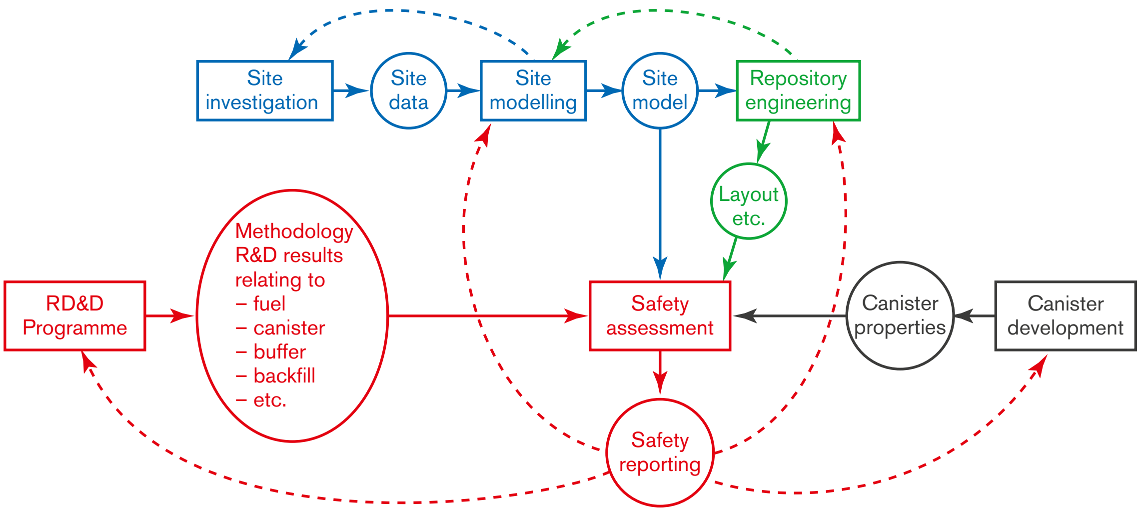


Figure 1-4. *Relations to other projects. Activities are shown as rectangles and products as circles or ellipses. As indicated by the dashed lines, the safety report provides feedback to repository engineering concerning e.g. layout issues and choice of backfill materials and to site investigations, via the site model, concerning detailed site investigation needs. The latter type of feedback is also given by the site modelling group independent of the safety assessment. Feedback is also given from the safety assessment to SKB's RD&D Programme.*

provides thermal, hydraulic, mechanical, chemical and transport properties of the rock, within a geometrical and geological framework describing major structures at the site. The biosphere part of the model includes a description of the ecosystems at the site and is developed to be coherent with the geosphere model. The site descriptive model is accompanied by a comprehensive description of the inter-disciplinary analysis and interpretation work underpinning it.

The site descriptive model of the Forsmark site provides descriptions of the present geosphere and biosphere conditions for the safety assessment. A more detailed account is given in Chapter 4.

The model describes the situation prior to rock excavation for the final repository. Analyses of how the excavation activities will affect the undisturbed, natural state of the rock are also needed. Parts of this work were undertaken by a repository engineering group, using the site description and in cooperation with the site model experts, in conjunction with their determination of a suitable repository layout in the site model. The results of these analyses are part of the input to SR-Site and are described in relevant parts of this report, in particular in Section 5.2. Additional analyses of the mechanical, hydrogeological and chemical evolution due to the excavation and operational period are made as part of SR-Site and are presented in Section 10.2.

Apart from providing descriptions of the geosphere and the biosphere, the site descriptive model gives an understanding of past and ongoing processes at the site. This information is useful for the description and modelling of the future development of the site and repository, the results of which have to be compatible with the understanding of the site history.

The safety assessment uses the hydrogeological simulation models set up by the site analysis group. Whereas these are essentially used to simulate the site history by the site analysis group, the future evolution is in focus in the safety assessment.

Data for SR-Site provided by the site model are, in the SR-Site project, assessed according to established procedures for the handling of all input data and data uncertainties, see further Chapter 9. The assessment of confidence in the site descriptive model made by the site modelling group as part of the documentation of the model provides important input to the data assessment in SR-Site.

The results of the safety assessment provide input to the planning of detailed site investigations in connection with repository excavation and to further design developments.

1.6.2 Repository engineering

A repository engineering group has developed a reference repository concept that is practically achievable while providing the required safety functions. The reference concept includes basic dimensions of the facilities as well as reference technical solutions for buffer and backfill. Using the reference concept and based on the site description, repository engineering then developed site-adapted layouts of the final repository. During the work, feedback was given to the continued site modelling and site investigation work. For the needs of SR-Site and SKB's licence application, the reference concepts, the site adapted repository layout and the methods for achieving these are documented in a number of so called **Production reports**, as further described in Chapter 5.

1.6.3 Canister development

Within the continuation of that project, techniques for canister production and sealing are further developed and documented. The project provides input to SR-Site in terms of canister properties. For the needs of SR-Site and SKB's licence application, the reference canister and the production methods for achieving it are documented in the **Canister production report**, see further Section 5.4.

2 Methodology

2.1 Introduction

This chapter outlines the methodology that has been used for SR-Site. The methodology builds on that presented in SKB's most recent comprehensive safety assessment the SR-Can study /SKB 2006a/. The developments since the SR-Can assessment are largely influenced by the findings in the review of SR-Can, in particular as expressed in the Swedish regulator's joint review report /Dverstorp and Strömberg 2008/. All central methodological elements are explained in Section 2.5 in sufficient detail to provide a self contained overview of the assessment methodology.

In broad terms, the methodology development for the SR-Can assessment was also influenced and inspired by several safety assessment studies in e.g. Switzerland /Nagra 2002/, Finland /Vieno and Nordman 1999/, Belgium /ONDRAF/NIRAS 2001/, Japan /JNC 2000/, the U.S. /BSC 2003/, Canada /Gierszewski et al. 2004/ and France /Andra 2005/ and by international cooperation in the area organised by the OECD Nuclear Energy Agency /NEA 1997a, 1999a, 2001, 2004a, b, 2009/.

The main purpose of a safety assessment of a final repository is to investigate whether the repository can be considered radiologically safe over time. In principle, this is established by comparing estimated releases of repository derived radionuclides and associated radiation doses with regulatory criteria (see Section 1.4 and Appendix A). For a KBS-3 repository, the primary safety function is to completely contain the waste for hundreds of thousands of years, see further Section 2.4. An important purpose of this safety assessment is, therefore, also to analyse the repository's potential for containing the wastes under a wide range of circumstances and for a very long time.

Appropriate scientific and technical support for all statements made and data selected is essential to give confidence in the calculated results. Demonstrating understanding of the disposal system and its evolution is thus a crucial component in any safety assessment.

The repository system, broadly defined as the deposited spent nuclear fuel, the engineered barriers surrounding it, the host rock and the biosphere in the proximity of the repository, will evolve over time. Future states of the system will depend on:

- its initial state,
- internal processes, i.e. a number of radiation related, thermal, hydraulic, mechanical, chemical and biological processes acting internally in the repository system over time, and
- external factors acting on the system.

Internal processes are e.g. the decay of radioactive material, leading to the release of heat and the subsequent warming of the fuel, the engineered barriers and the host rock. Groundwater movements and chemical processes affecting the engineered barriers and the composition of groundwater are other examples. External factors include effects of future climate and climate-related processes, such as glaciations and land uplift. Another example is the build-up of mechanical energy due to plate tectonic movements. Also, future human actions may influence the repository.

The initial state, the internal processes and the external influences and the way they together determine repository evolution, can never be fully described or understood. There are thus uncertainties of various types associated with all aspects of the repository evolution and hence with the evaluation of safety. A central theme in any safety assessment methodology must therefore be the management of all relevant types of uncertainty. This management amounts to identifying, classifying and describing uncertainties, as well as handling them in a consistent manner in the quantification of the repository evolution and of the radiological consequences to which it leads. It also implies comparing the results of the assessment with regulatory criteria in such a way that appropriate allowance is made for the uncertainties associated with the assessment.

The primary safety function of the KBS-3 system described in Figure 1-1 is to completely contain the spent nuclear fuel within the copper/iron canisters over the entire assessment period. Should a canister be damaged, the secondary safety function is to ensure that any releases from the canister are retarded sufficiently to ensure that the resultant radionuclide concentrations are reduced to levels that do not cause unacceptable consequences. The two issues of containment and retardation are, therefore, principal considerations throughout the assessment.

In the next Section 2.2, safety principles for the KBS-3 repository are discussed in more detail. Thereafter the boundary of the repository system is established in Section 2.3 and the relevant time scales for the assessment are given in Section 2.4.

With these fundamentals established, an account of the methodology for the SR-Site assessment is given in Section 2.5. This section is considerably extended compared to the corresponding section in the SR-Can main report, but some details of the methodology are still best explained as they are applied, meaning that the reader is sometimes referred to subsequent chapters for details.

Methodological aspects of the risk calculation are addressed in Section 2.6.

The safety assessment has an important role in the demonstration of use of best available technique, BAT. This role and how it is fulfilled is discussed in Section 2.7.

The handling of uncertainties permeates the safety assessment and is thus an integral part of the methodology described in Section 2.5. Section 2.8 gives a more thorough account of the identification, classification and handling of uncertainties in the SR-Site assessment. This is important since a rigorous handling of uncertainties is closely related to the confidence in the results of the assessment.

Also, quality assurance measures concern all aspects of the assessment. The quality assurance plan for SR-Site is presented in Section 2.9.

2.2 Safety

2.2.1 Safety principles for the KBS-3 repository

Since work on the Swedish final repository project commenced at the end of the 1970s, SKB has established a number of principles for the design of a final repository. The principles can be said to constitute the safety philosophy behind the KBS-3 concept. They are summarised below.

- By placing the repository at depth in a long-term stable geological environment, the waste is isolated from the human and near-surface environment. This means that the repository is not strongly affected by either societal changes or the direct effects of long-term climate change at the ground surface.
- By locating the repository at a site where the host rock can be assumed to be of no economic interest to future generations, the risk of human intrusion is reduced.
- The spent fuel is surrounded by several engineered and natural safety barriers.
- The primary safety function of the barriers is to contain the fuel within a canister.
- Should containment be breached, the secondary safety function of the barriers is to retard a potential release from the repository.
- Engineered barriers shall be made of naturally occurring materials that are stable in the long term in the repository environment.
- The repository shall be designed and constructed so that temperatures that could have significant detrimental effects on the long-term properties of the barriers are avoided.
- The repository shall be designed and constructed so that radiation induced processes that could have significant detrimental effects on the long term behaviour of the engineered barriers or of the rock are avoided.
- The barriers should be passive, i.e. they should function without human intervention and without artificial supply of matter or energy.

Together with many other considerations, like the geological setting in Sweden and the requirement that the repository must be feasible to construct from a technical point of view, these principles have led to the development of the KBS-3 system for spent nuclear fuel management.

2.2.2 Safety functions and measures of safety

The key safety related features of the KBS-3 disposal system can be summarised in the safety functions containment and retardation.

The fuel is placed in corrosion-resistant copper canisters with a cast iron insert providing mechanical strength. The copper canisters are surrounded by bentonite clay in deposition holes at a depth of approximately 500 m in the host rock (see Figure 1-1). The bentonite clay protects the canisters from minor rock movements and limits the inflow of the low concentrations of corrosive agents in the groundwater. The host rock provides a long-term chemically, mechanically, thermally and hydrogeologically stable environment for the canisters and the bentonite clay. The canisters, therefore, constitute a containing barrier with a very long life-time in the environment provided by the buffer and the host rock.

The fuel, the canister, the buffer and the host rock contribute to retarding any potential release of radionuclides should a canister be damaged. The fuel matrix is in itself very stable in the reducing environment at repository depth. Many of the most hazardous radionuclides have a very low solubility in groundwater and thereby have a limited potential for outward transport. Both the cast iron insert and the copper canister limit the inflow of water even if damaged. The buffer limits the inflow of water to a damaged canister. It also limits the release of radionuclides by limiting water flow and through sorption. The groundwater moves slowly in the fracture system of the rock nearest to the canisters and many radionuclides have a strong propensity for diffusion into, and sorption in, the host rock matrix.

The fundamental criterion regarding safety is expressed in SSM's regulation SSMFS 2008:37 where it is stated that the aim is to ensure that "the annual risk of harmful effects after closure does not exceed 10^{-6} for a representative individual in the group exposed to the greatest risk". The risk criterion corresponds to effective doses that are roughly one percent of those due to naturally occurring background radiation. As mentioned in Section 1.4.1, effects on living environments and ecosystems shall also be described.

Results of risk calculations in the safety assessment are compared with this criterion in order to assess compliance. However, the risk results depend in a complex fashion on a large number of factors. In a safety assessment, it is necessary to not only assess compliance with an overall criterion, but also to demonstrate how safety is related to key properties of the barriers and how these properties vary over time. An obvious property is the integrity of the copper canisters. This in turn depends on a number of factors like the buffer properties and the chemical environment of the repository.

In Chapter 8, a number of safety function indicators for the barriers are presented and discussed. Criteria are provided for properties like buffer temperature and buffer and backfill density, hydraulic conductivity and swelling pressure. Demonstrating compliance with these criteria provides arguments that the barriers will function as intended as the repository system evolves. Conversely, should a safety function indicator criterion be breached, this signals that safety in one way or the other is potentially jeopardised and that the consequences need to be further considered. However, it does not automatically imply that overall system performance is unacceptable.

Section 2.6.3 provides a discussion of alternative "top level" indicators to the risk criterion. The alternative criteria are more directly related to releases from the geosphere and do not require detailed assumptions about biosphere conditions or human habits.

2.3 System boundary

The repository system encompasses the spent nuclear fuel, the canisters, the buffer, the backfilled deposition tunnel and other repository cavities, the geosphere and the biosphere in the proximity of the repository, see Figure 1-1. In the development of the SR-Can database of features, events and processes, FEPs, (see below), the system boundary was defined in more detail. The following key aspects are taken from that definition.

- In general, a strict boundary definition is neither possible nor necessary, and the same boundaries will not necessarily be relevant to all parts of the safety assessment. The following definitions were the basis for the FEP sorting – and thus have affected the system description.

- Roughly, the portion of the biosphere studied in the site investigations, i.e. an area of the order of 100–300 km^2 above the repository, is regarded as part of the system, whereas the biosphere on a larger scale is regarded as external. The analysis of the biosphere extends downward to the rock surface in this assessment. Depending on the analysis context this definition may be somewhat modified.
- Roughly the corresponding portion of the geosphere down to a depth of about 1,000 m is regarded as part of the system. Depending on the analysis context, this definition may also be modified. For example, the local groundwater model, which is the scale most relevant to safety, has a smaller projected surface area than 100 km^2, whereas e.g. larger areas than 300 km^2 and greater depths than 1,000 m may be required for regional groundwater modelling. Boundary conditions for the local groundwater model are provided from such a larger regional model.
- Future human behaviour on a local scale is internal to the system, but not issues related to the characteristics and behaviour of future society at large.

2.4 Timescales

A timescale for the safety assessment needs to be established since this provides a general limit on the scope of the assessment and also cut-off times for e.g. radionuclide transport calculations. The issue is addressed in applicable regulations as cited below.

2.4.1 Regulatory requirements and guidance

The SSM regulations SSMFS 2008:21 state that the safety assessment should cover the period during which the barrier functions are needed, though to at least 10,000 years after closure. The recommendations accompanying the regulation suggest that the timescale of an assessment should be related to the hazard posed by the inventory in comparison with naturally occurring radionuclides. In the recommendations it is also noted that "…it should also be possible to take into consideration the difficulties of conducting meaningful analyses for extremely long time-periods, beyond one million years…".

SSM's regulation 2008:37 state that "For the first thousand years following repository closure, the assessment of the repository's protective capability shall be based on quantitative analyses of the impact on human health and the environment." "For the period after the first thousand years following repository closure, the assessment of the repository's protective capability shall be based on various possible sequences for the development of the repository's properties, its environment and the biosphere."

The General Guidance to SSMFS 2008:37 states the following regarding a repository for spent nuclear fuel: "…the risk analysis should at least include approximately one hundred thousand years or the period for a glaciation cycle to illustrate reasonably predictable external strains on the repository. The risk analysis should thereafter be extended in time as long as it provides important information about the possibility of improving the protective capability of the repository, although at the longest for a time period of up to one million years".

Regarding the quantitative risk analysis for the first 100,000 years the General Guidance state the following: "Supplementary indicators of the repository's protective capability, such as barrier functions, radionuclide fluxes and concentrations in the environment, should be used to strengthen the confidence in the calculated risks."

For the time beyond approximately one hundred thousand years, the guidance furthermore state: "A strict quantitative comparison of calculated risk in relation to the criterion for individual risk in the regulations is not meaningful. The assessment of the protective capability of the repository should instead be based on reasoning on the calculated risk together with several supplementary indicators of the protective capability of the repository such as barrier functions, radionuclide fluxes and concentrations in the environment."

2.4.2 Timescale covered by the safety assessment

Apart from the obvious necessity of fulfilling regulatory requirements, arguments relating to the radiotoxicity of the spent nuclear fuel can also be considered when a timescale for a safety assessment is determined.

After approximately 100,000 years, the radiotoxicity of the spent nuclear fuel is comparable with that of the natural uranium ore once used to produce the fuel /Hedin 1997/. Also the sum of toxicity of all fractions originating from the nuclear fuel cycle (the daughter nuclides separated from the uranium prior to enrichment, the depleted uranium arising in the enrichment process and the spent fuel) is comparable to that of the utilised uranium ore after 100,000 years, see Figure 2-1. The latter comparison is equivalent to comparing the radiotoxicity of the amount of natural U-235 and U-238 consumed in the reactor, to the radiotoxicity of the amounts of the new products created in the reactor (fission products and actinides) remaining after 100,000 years.

Another criterion that may be considered to justify a timescale for a safety assessment is that the period analysed should go beyond the point in time at which peak doses from the repository occur. However, on long timescales, the peak dose often occurs at the end of the typically one million year assessment period due to in-growth of the naturally occurring nuclide Ra-226 from disposed U-238 combined with the fact that the barrier system inevitably deteriorates over time. This was the case in the SR-Can assessment. Since the KBS-3 concept is aiming at complete containment of the waste for time periods very far into the future through encapsulation, the peak dose criterion is not considered an appropriate criterion for defining the assessment timescale.

In SR-Site, the timescale for the assessment is one million years. This timescale is in accordance with the suggestions in the recommendations and general guidance cited above. It is furthermore longer than that needed to reduce the radiotoxicity of the inventory to a level comparable with that of the corresponding amount of natural uranium ore. It is also noted that the key radionuclides remaining in the waste beyond one million years are also those, such as U-238, associated with natural uranium ore.

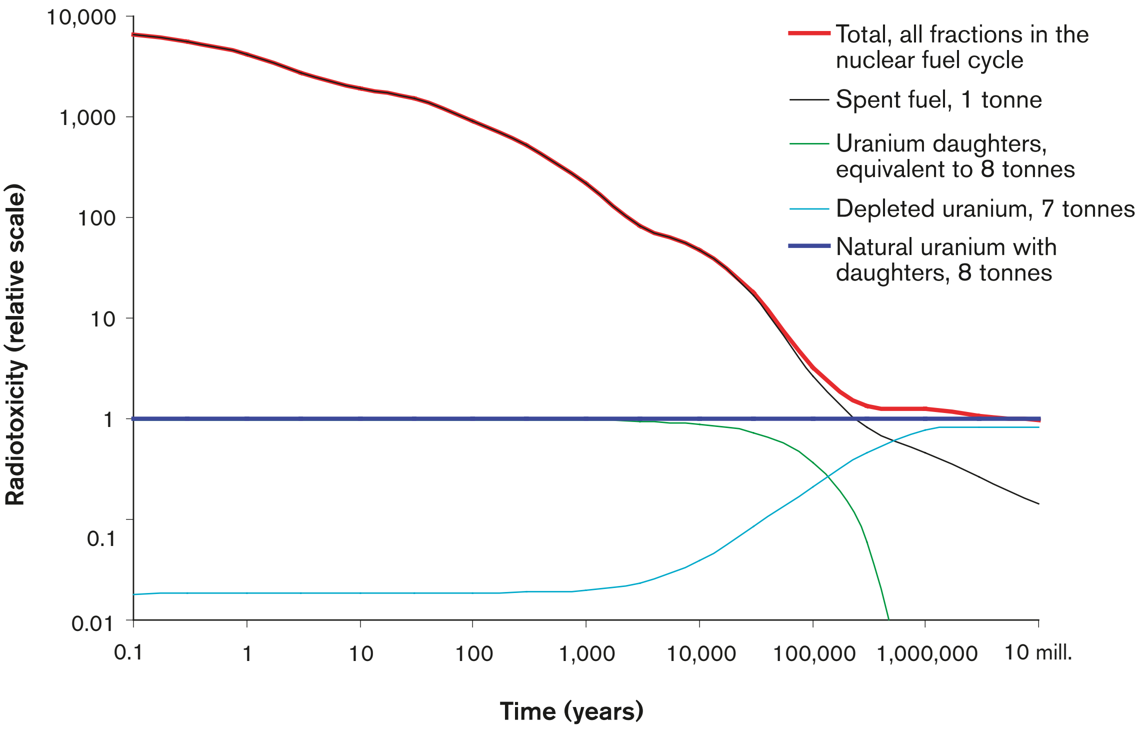


Figure 2-1. *Radiotoxicity on ingestion of uranium and daughters in ore (blue line), and of the sum of all fractions that arise when the same quantity of uranium is used in the nuclear fuel cycle (red line). The time refers to the time after reactor operation. The different fractions comprise the spent fuel (38 MWd thermal energy/kg U of type SVEA 64 BWR), the depleted uranium and the uranium daughters that are separated in the uranium mill. From /Hedin 1997/.*

As expressed in the guidance to SSMFS 2008:37, the quantitative risk criterion is applicable as a quantitative regulatory limit during approximately the first one hundred thousand years, and thereafter as a basis for discussing the protective capability of the repository. The risk calculations in SR-Site therefore extend to one million years, and the results are used in accordance with SSM's General Guidance in the compliance discussion in Chapter 15. The risk calculations are complemented with calculations of supplementary indicators of repository safety, see further Section 2.6.3.

However, a brief general discussion of the evolution beyond one million years is also given, to demonstrate the effects of a continued development of important processes in the repository and also since a strict cut-off time for the analysis of the general evolution of the system cannot be derived from the regulations.

Regarding the regulatory requirement of a more detailed analysis during the initial 1,000 years after closure, motivated both by the high radiotoxicity of the spent fuel during this period and the better ability to predict this relatively near future, this is partly fulfilled as the initial, transient stages of the repository evolution are analysed, mainly in the reference evolution described in Chapter 10. To fully fulfil the regulatory requirement of a more detailed analysis for the initial 1,000 years, the functioning of the barrier system during this period is revisited later in the report, covering also hypothetical early barrier failure, see Section 13.9.5.

For times beyond one million years, a qualitative account of the repository evolution, based on the quantitative analyses up to one million years, is provided. No risk calculation is presented for these time scales.

2.4.3 Timescales relevant for repository evolution

There are a number of timescales relevant to repository evolution as summarised below.

- A fundamental timescale is that relevant for the decrease of the radiotoxicity of the waste as shown in Figure 2-1. At the time of deposition, the radiotoxicity has decreased by roughly a factor of ten compared with the situation one month after reactor operation, and then continues to decrease by about a factor of ten for every ten-fold increase in time. As mentioned above, after 100,000 years the radiotoxicity of a certain amount of spent nuclear fuel is comparable with the radiotoxicity of the amount of natural uranium ore once used to produce the fuel.
- It is also noted that the initially very high dose rates from potential exposure to direct, external radiation from the spent fuel decrease substantially within a few thousand years. In the long term, these dose rates will however remain at levels requiring shielding from humans practically indefinitely, since the long-term direct radiation level is determined by U-238 progeny. For long-term safety, direct radiation to humans is only a concern in scenarios addressing unintentional intrusion into the repository. (Processes relating to radiation effects in an undisturbed repository are included in the analysis of repository evolution.)
- The timescale of long-term geological processes, occurring over millions of years, including tectonic movements of continental plates and associated ridge push caused by these movements.
- Climate change occurs on timescales of a few tens of years up to more than one million years. One main timescale relate to the length of glacial cycles, which for the past approximately 700,000 years evolved in ~100,000 year cycles. In the region where Forsmark is now located each cycle includes several episodes of permafrost and glacial conditions. The mechanical, hydraulic and groundwater chemical conditions in the host rock vary in consequence of the climatic evolution, in particular as a result of glacial overriding. It is possible that these cycles are or will be perturbed by human-induced climate changes, but the amount and persistence of such perturbations remains a subject of investigation.
- There are a number of timescales on which biological evolution occurs; e.g. man has evolved considerably during the past several hundred thousand years.
- The natural development of ecosystems in general could lead to considerable changes in a 1,000 year perspective. This is e.g. the case for coastal ecosystems in Sweden, which are strongly affected by land-uplift.

- Most aspects of society have changed substantially over the past 100 years and significant changes may occur abruptly or over only a few years. Historical records of humanity cover a few thousand years.
- The residual power of the fuel results in peak temperatures in the near field of the repository after of the order of ten years, and elevated temperatures in the host rock for a few thousand years.
- The resaturation of the buffer, the backfill and the host rock typically requires tens to hundreds of years for Swedish conditions.
- The chemical conditions in the host rock after excavation and operation of a final repository are expected to have largely returned to natural conditions in a 100 or 1,000 year perspective. The chemical conditions in the buffer will change to some degree during the period of elevated temperatures. Canister corrosion under typical repository conditions requires millions of years to cause canister failures.

Timescales is a recurring issue in this report, e.g. in the context of process descriptions, Section 7.3, and when analysing repository evolution, Chapter 10. The issue of timescales in safety assessments has also been addressed in an NEA Workshop /NEA 2004a/ in which SKB participated. Approaches used in SR-Site regarding e.g. the focussing on complementary indicators to dose and risk for long time scales and the use of scenarios to cover a range of possible evolutions of future external conditions are in agreement with the findings of the NEA workshop.

2.5 Methodology in eleven steps

The safety assessment SR-Site consists of eleven main steps, which are carried out partly concurrently and partly consecutively. From a project management point of view, many of the steps can be seen as sub-projects in a larger integrated safety assessment project. Figure 2-2 is a graphical illustration of the steps.

The methodology followed in the eleven steps of the assessment is described in the following subsections. Inevitably, some details of the methodology are best explained as they are applied and the reader is therefore sometimes referred to subsequent chapters for these details.

2.5.1 Step 1: FEP processing

This step consists of identifying all the factors that need to be included in the analysis. Experience from earlier safety assessments and KBS-3 specific and international databases of relevant features, events and processes (FEPs) influencing long-term safety are utilised. An SKB FEP database is developed where the great majority of FEPs are classified as being either i) initial state FEPs, ii) internal processes or iii) external FEPs. Remaining FEPs are either related to assessment methodology in general or determined to be irrelevant for the KBS-3 concept.

Based on the results of the FEP processing, an SR-Site FEP catalogue, containing FEPs to be handled in SR-Site, has been established.

The further handling of the three FEP categories is described in the three subsequent steps of the methodology.

This step of FEP processing is further described in Chapter 3 and fully documented in the SR-Site **FEP report**[4].

2.5.2 Step 2: Description of the initial state

The initial state of the system is described, based on the descriptive model of the repository site, the KBS-3 repository design with its different components and a site-specific layout applying this design

[4] The **FEP report** is one of several principal references in this main report. See Section 2.5.12 for a complete list and nomenclature for referencing in terms of short-names in bold.

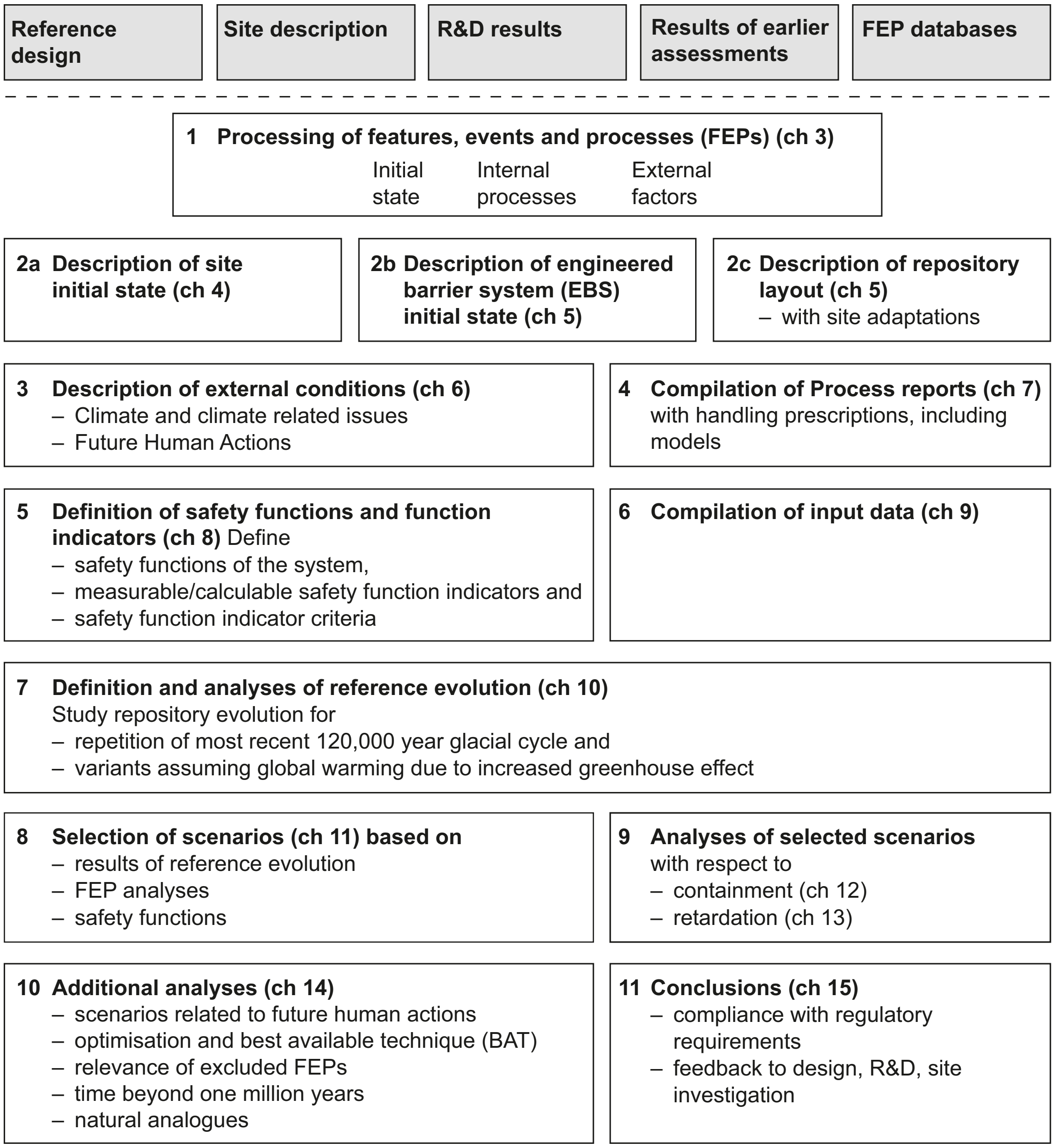


Figure 2-2. *An outline of the eleven main steps of the SR-Site safety assessment. The boxes at the top above the dashed line are inputs to the assessment. The products of each step are described in detail in the main text. Together, the eleven steps represent the box "Safety assessment" in Figure 1-4.*

to the site. The initial state of the geosphere and the biosphere is that of the natural system prior to excavation. The initial state of the fuel and the engineered components is that immediately after deposition. The choice of time for the initial state is further elaborated in Section 5.1.

The initial state of the system is a fundamental input to the assessment and needs thorough substantiation. For the site, this is provided by the site descriptive model of the Forsmark site /SKB 2008a/, i.e. the results of the surface based site investigation and the modelling of the site based on the site investigation data. The Forsmark site model is a fundamental reference to SR-Site and a summary of the site description, with a focus on aspects relevant for long-term safety is given in Chapter 4.

The initial state of engineered components of the repository system are described in a number of so called **Production reports** covering the spent fuel, the canister, the buffer, the tunnel backfill, the repository closure and the underground openings constructions, respectively. The last report contains a description of the repository layout after site adaptation. Each production report gives an account of

i) design premises derived from the SR-Can assessment, ii) the reference design selected to achieve the requirements, iii) verifying analyses that the reference design does fulfil the design premises, iv) the production and control procedures selected to achieve the reference design, v) verifying analyses that these procedures, if implemented, would achieve the reference design and vi) an account of the achieved initial state. The last point is the key input to the safety assessment. The initial state of the engineered components is described in Chapter 5, based on the contents of the **Production reports** relevant to long-term safety.

The FEP processing in step 2 resulted in a number of FEPs related to the initial state. Most of these are covered through the descriptions of the initial state and the subsequent use of this information in the assessment. A few initial state FEPs, relating for example to an incomplete closure of the repository require separate treatment in the scenario analysis in later stages of the assessment. Such FEPs are accounted for in Section 5.1.3.

2.5.3 Step 3: Description of external conditions

Factors related to external conditions are handled in the three categories “climate related issues”, “large-scale geological processes and effects” and “future human actions”, FHA. The handling of these factors is described in the **Climate report**, the **Geosphere process report**, and the **FHA report**, respectively.

A key point in the handling of external conditions is the establishment of reference external conditions for the subsequent analysis. These reference external conditions postulate a repetition of the last 120,000 year glacial cycle, the Weichselian. An alternative reference evolution is based on the assumption of a global warming effect. In addition, physically possible climate conditions that would have the most severe impact on repository safety are sought for use in the scenario analyses in a later step of the assessment. The handling of climate related issues is described in more detail in Section 6.2.

Future human actions are handled according to a methodology established in the SR-Can assessment with minor updates for SR-Site. Based on a structured account of a large number of FEPs relating to FHA, a selection of stylised cases for further analyses is made. This is described in the analysis of FHA scenarios, Section 14.2.

2.5.4 Step 4: Description of processes

The identification and handling of processes of importance for the long-term evolution and safety of the repository is a key element in the safety assessment. The identification of processes is based on earlier assessments and FEP screening. All identified processes within the system boundary relevant to the long-term evolution of the system are described in three dedicated **Process reports**, one for the fuel and canister, one for the buffer, backfill and repository closure and one for the geosphere. Short-term geosphere processes/alterations due to repository excavation are included in the **Geosphere process report** and are taken into account in the assessment.

Each process is documented in the **Process reports**, following a template with the following headings:

- Overview/general description.
- Dependencies between process and system variables.
- Boundary conditions.
- Model studies/experimental studies.
- Natural analogues/observations in nature.
- Time perspective in which the process is relevant.
- Handling in the safety assessment SR-Site.
- Handling of uncertainties in SR-Site.
- Adequacy of references supporting the handling in SR-Site.

The description of boundary conditions (second heading) points to interactions with processes in other process reports when relevant. The material under the last two headings but one, where it is established how the process will be handled in the safety assessment, constitutes the key output from the **Process reports**. The **Process reports** thus provide a "recipe" for the handling of the various processes in the assessment.

The handling of all processes in a process report is summarised in a process table describing if the process is neglected, if it is quantitatively modelled or if the choice between neglect and modelling is subject to a specified condition that may or may not be fulfilled as the repository system evolves.

Several of the processes are thus handled through quantitative modelling, where each model in general includes several interacting processes, often occurring in different system parts and hence described in different process reports. The models form a network, where results from one model are used as input to another. The network is described graphically by two Assessment Model Flowcharts, AMFs, and two associated AMF tables linking the processes in the process tables, the models in the AMFs and the reporting of the modelling in this main report.

The handling of processes is described in more detail in Chapter 7, where the process tables (Section 7.4), the AMFs and the AMF tables (Section 7.5) are also provided.

2.5.5 Step 5: Definition of safety functions, safety function indicators and safety function indicator criteria

A central element in the methodology of the SR-Site assessment is the definition of a set of *safety functions* that the repository system should ideally fulfil over time. Here, the overall safety functions containment and retardation are differentiated into a number of lower level functions for the canister, the buffer, the deposition tunnel backfill and the host rock. The evaluation of the safety functions over time is made possible by associating every safety function with a *safety function indicator*, i.e. a measurable or calculable property of the repository component in question. For several functions, it is also possible to associate a *safety function indicator criterion* such that if the safety function indicator fulfils the criterion, then the safety function in question is upheld.

The ability of the canister to resist isostatic load is an example of a safety function. The associated indicator is the isostatic load on the canister and the criterion is the isostatic load that the canister has been demonstrated to sustain.

All safety functions, indicators and criteria are provided in Chapter 8 where a more elaborated description of the methodology behind the safety functions is also given. Detailed safety functions for containment and retardation are given in Sections 8.3 and 8.4, respectively.

It is important to note that the safety function indicator criteria are not the same as design criteria, formally described as design premises in SR-Site. The former should ideally be upheld throughout the assessment period whereas the design premises should be fulfilled initially. In general, the design premises should assure that the system is robust to the extent that the safety functions indicator criteria are fulfilled over time. For example, the copper canister must be designed such that its initial thickness (the design premise) ensures that it will sustain corrosion for a very long time, i.e. such that the thickness is non-zero (the function indicator criterion) during this time.

The set of safety functions in itself provides understanding of the safety features of the system and a list of key issues to evaluate over time in the assessment. The safety functions are explicitly used in later steps of the assessment to evaluate safety in a structured manner over the different time frames when analysing a reference evolution. A key use of the safety functions is their role in the selection of a number of scenarios whereby uncertainties related to the safety of the system are evaluated in a structured manner, see further Section 2.5.8.

When the safety functions are defined, a FEP chart is developed, showing how initial state factors and processes in the long-term evolution of the repository are related to the safety functions.

2.5.6 Step 6: Compilation of data

In this step, data to be used in the quantification of repository evolution and in dose calculations are selected using a structured procedure. The process of selection and the data values adopted are reported in a dedicated **Data report**. The process follows a template for discussion of input data uncertainties. The template is given in Chapter 9 and the selected data are provided in the **Data report**. The models for which data are required are given in the AMF described in step 4 (Section 2.5.4).

Compared to the SR-Can assessment the structured compilation of data has been extended to more completely cover the calculations in the assessment, see Section 9.3. Also, the template has been updated in response to findings in the review of the SR-Can assessment.

2.5.7 Step 7: Analysis of reference evolution

In this step, a reference evolution of the repository system that follows from the reference external conditions defined in step 3 is defined and analysed. The purpose is to gain an understanding of the overall evolution of the system and of uncertainties affecting the evolution, for the scenario selection and scenario analyses that follow in the two subsequent steps. The evolution is an important basis for the later definition of a main scenario, see Section 2.5.8.

Focus is on the containment capacity of the system. Two cases of the reference evolution are analysed.

1. A base case in which the external conditions during the first 120,000 year glacial cycle are assumed to be similar to those experienced during the most recent cycle, the Weichselian. Thereafter, seven repetitions of that cycle are assumed to cover the entire 1,000,000 year assessment period.
2. A global warming variant in which the future climate and hence external conditions are assumed to be substantially influenced by human-induced greenhouse gas emissions. This analysis is related to that of the base case.

For both these, the initial state with its uncertainties described in Chapter 5 is assumed, all internal processes, with their uncertainties, are handled according to the specification given in the **Process reports**, as summarised in Chapter 7 and data with their uncertainties are taken from the **Data report** as summarised in Chapter 9.

The presentation of the analysis of the base case of the reference evolution is divided into four time frames:

- The excavation/operational period.
- The first 1,000 years after repository closure and the initial period of temperate domain from the reference glacial cycle.
- The remaining part of the glacial cycle.
- Subsequent glacial cycles up to one million years after repository closure.

For each time frame, issues are presented in the following order:

- Climate issues.
- Biosphere issues.
- Thermal, mechanical, hydraulic and chemical issues in the geosphere.
- Thermal, mechanical, hydraulic and chemical issues for the engineered barrier system (canister, buffer and backfill).

The discussion of each of the issues is concluded with an account of identified uncertainties to be propagated to later stages of the reference evolution and to subsequent parts of the safety assessment.

The commentary on each time frame concludes with a discussion of the expected status of the safety function indicators defined in Chapter 8 during and at the end of the time frame.

A considerable part of the material presented is results from simulation studies, mentioned in step 4 (Section 2.5.4) and graphically represented in the assessment model flow charts, AMFs.

After the analysis of the base case, the global warming variant is analysed under the headings “External conditions”, “Biosphere”, “Repository evolution” and “Safety functions”, essentially as comparisons to analyses of the base case.

2.5.8 Step 8: Selection of scenarios

A key feature in managing uncertainties in the future evolution of the repository system is the reduction of the number of possible evolutions to analyse by selecting a set of representative scenarios. The selection focuses on addressing the safety relevant aspects of the evolution expressed at a high level by the safety functions 'containment' and 'retardation' which are further characterised by reference to safety function indicators in Chapter 8.

The selected scenarios should cover all reasonable future evolutions. Furthermore, it should be possible to logically calculate the risk associated with the presence of the repository as a sum of risk contributions from the set of scenarios, as discussed further in Section 2.6.2, subheading "Scenario disaggregation".

Regulatory requirements and recommendations

There are several issues concerning applicable regulations that have to be taken into account in the selection of scenarios. The quantitative criterion for repository safety in Swedish regulations is a risk limit and from the analyses of the defined scenarios it must therefore be possible to draw conclusions regarding risk.

SSM's Regulations SSMFS 2008:21 require that scenarios be used to describe future potential evolutions of the repository and that among these, there should be a main scenario that takes into account the most likely changes within the repository and its surroundings.

The General Recommendations concerning SSMFS 2008:21 describe a scenario in the safety assessment as comprising "a description of how a given combination of external and internal conditions affect repository performance".

The General Recommendations describe three types of scenarios: the main scenario, which includes the expected evolution of the repository system; less probable scenarios, which include alternative sequences of events to the main scenario and also the effects of additional events; and residual scenarios, which evaluate specific events and conditions to illustrate the function of individual barriers. For these categories SSM's Recommendations state the following:

"The main scenario should be based on the probable evolution of external conditions and realistic, or where justified, pessimistic assumptions with respect to the internal conditions. It should comprise future external events which have a significant probability of occurrence or which cannot be shown to have a low probability of occurrence during the time covered in the safety assessment. Furthermore, it should be based, as far as possible, on credible assumptions with respect to internal conditions, including substantiated assumptions concerning the occurrence of manufacturing defects and other imperfections, and which allow for an analysis of the repository barrier functions (it is, for example, not sufficient to always base the analysis on leak-tight waste containers, even if this can be shown to be the most probable case). The main scenario should be used as the starting point for an analysis of the impact of uncertainties (see below), which means that the analysis of the main scenario also includes a number of calculation cases.

Less probable scenarios should be prepared for the evaluation of scenario uncertainty (see also below). This includes variations on the main scenario with alternative sequences of events as well as scenarios that take into account the impact of future human activities such as damage inflicted on barriers. (Damage to humans intruding into the repository is illustrated by residual scenarios, see below). The analysis of less probable scenarios should include analyses of such uncertainties that are not evaluated within the framework of the main scenario.

Residual scenarios should include sequences of events and conditions that are selected and studied independently of probabilities in order to, inter alia, illustrate the significance of individual barriers and barrier functions. The residual scenarios should also include cases to illustrate damage to humans intruding into the repository as well as cases to illustrate the consequences of an unclosed repository that is not monitored."

Regarding scenario probabilities, the SSM Recommendations state: "The probabilities that the scenarios and calculation cases will actually occur should be estimated as far as possible in order to calculate risk."

SSM's General Guidance on application of SSMFS 2008:37 defines a scenario as a "description of the development of the repository given an initial state and specified conditions in the environment and their development".

Regarding the choice of scenarios, SSM's General Guidance states the following:

"The assessment of the protective capability of the repository and the environmental consequences should be based on a set of scenarios that together illustrate the most important courses of development of the repository, its surroundings and the biosphere.

Taking into consideration the great uncertainties associated with the assumptions on climate evolution in a remote future and to facilitate the interpretation of the risk to be calculated, the risk analysis should be simplified to include a few possible climate evolutions.

A realistic set of biosphere conditions should be associated with each climate evolution. The different climate evolutions should be selected so that they together illustrate the most important and reasonably foreseeable sequences of future climate states and their impact on the protective capability of the repository and the environmental consequences."

"The risk from the repository should be calculated for each assumed climate evolution by summing the risk contributions from a number of scenarios that together illustrate how the more or less probable courses of development in the repository and the surrounding rock affect the repository's protective capability and environmental consequences. The calculated risk should be reported and evaluated in relation to the criterion of the regulations for individual risk, separately for each climate evolution."

SSM's General Guidance states: "A number of scenarios for inadvertent human impact on the repository should be presented. The scenarios should include a case of direct intrusion in connection with drilling in the repository and some examples of other activities that indirectly lead to a deterioration in the protective capability of the repository, for example by changing groundwater chemistry or the hydrological conditions in the repository or its surroundings. The selection of intrusion scenarios should be based on present living habits and technical prerequisites and take into consideration the repository's properties.

The consequences of the disturbance of the repository's protective capability should be illustrated by calculations of the doses for individuals in the most exposed group, and reported separately apart from the risk analysis for the undisturbed repository. The results should be used to illustrate conceivable countermeasures and to provide a basis for the application of best available technique".

SSM's General Guidance to SSMFS 2008:37 states the following: "An account need not be given of the direct consequences for the individuals intruding into the repository." It is noted that this is contrary to the view expressed in SSMFS2008:21, where these situations are included among the residual scenarios.

The guidance to SSMFS 2008:37 also mention "special scenarios": "… an analysis of a conceivable loss, during the first thousand years after closure, of one or more barrier functions of key importance for the protective capability should be made separately from the risk analysis. The intention of this analysis should be to clarify how the different barriers contribute to the protective capability of the repository."

Method for scenario selection

Given the regulatory requirements and the general considerations discussed above, a method for the selection of scenarios in five steps has been developed as explained below.

1. Definition of the main scenario

A main scenario is defined, based on the reference evolution and in accordance with SSMFS 2008:21. The main scenario is split into two variants, based on the two variants of the reference evolution, (the Weichselian base case and the global warming variant). (The reference evolution corresponding to the main scenario is explained in general terms in step 7, and the handling of uncertainties within the reference evolution is further developed in Section 2.8.4.)

(It is noted that SSM uses "climate evolutions" as a hierarchical level above the scenarios. In order to not complicate the description further, the different types of climate evolution mentioned in SSM's General Guidance are handled as variants of scenarios.)

2. Selection of additional scenarios based on potential loss of safety functions

A main factor governing scenario selection is the concern that the intended safety functions relating to containment (Chapter 8) should be upheld. Therefore, these safety functions are used to structure the selection of additional scenarios. This is the main approach for addressing the issue of less probable scenarios, in relation to SSMFS 2008:21.

There are three canister safety functions related to containment: to provide a corrosion barrier, to withstand isostatic load and to withstand shear load. Three distinct canister failure modes, due to corrosion, isostatic pressure and shear movement, respectively, can thus be derived from the safety functions. Therefore, three scenarios, one for each canister failure mode, are generated. Three 'failed' states of the buffer; advective, frozen and transformed, are also considered as scenarios. The canister scenarios are systematically combined with the buffer scenarios.

For each selected scenario, uncertainties related to initial state factors, processes and external conditions that are not covered in the main scenario are considered. In e.g. the case of canister failure due to isostatic overpressure, inadequacies in the manufacturing of the load-bearing canister insert, higher than reference buffer swelling pressures and extreme ice sheets yielding high groundwater pressures are considered.

The FEP chart, see step 5 (Section 2.5.5), aids in ensuring that all conceivable routes to deficiencies in containment are captured. Systematic analyses of initial state factors, long-term processes and external conditions possibly contributing to each of these scenarios are made. The results of the analysis of the main scenario, with all the coupled FEPs and uncertainties considered there, is an important starting point for this assessment.

Based on this information, an assessment of whether each scenario is to be considered as "less probable" or "residual" is made. In the former case, the likelihood of the scenario is normally pessimistically set to one, whereas the assessed limited likelihoods of its characteristic FEPs, e.g. large earthquakes, are taken into account in the risk calculation associated with the scenario.

These scenarios also cover many of the residual scenarios required by SSM's Regulations and General Guidance to analyse the significance of barriers and barrier functions. To obtain a deeper understanding of barrier functions, a number of residual scenarios are defined illustrating, from the point-of-view of radionuclide transport, hypothetical situations where one or several barriers are assumed to be initially lost, see further Section 13.7.3.

The selection of additional scenarios is described in detail in Section 11.2.

The analyses of the selected additional scenarios are described in Chapter 12, from Section 12.2 onward.

3. Scenarios related to future human actions

A set of scenarios related to future human actions was also defined and analysed. Human intrusion scenarios resulting in a degradation of system performance are to be considered as "less probable scenarios" according to SSMFS 2008:21, but not included in the risk summation according to the General Guidance to SSMFS 2008:37. SSM requires residual scenarios to illustrate damage to humans intruding into the repository and cases to illustrate the consequences of an unclosed repository that is not monitored.

The selection and consequence analyses of scenarios related to future human actions are reported in Section 14.2, as part of the documentation of additional analyses in Chapter 14. The FHA scenarios are reported separately from the main scenario and the additional scenarios based on potential loss of safety functions, since the methodology for the definition of FHA scenarios as well as the analyses of containment and retardation differs from the approaches for other scenarios.

4. Other residual scenarios, etc.

Any other scenarios that are, for any reason, considered necessary in order to obtain an adequate set of scenarios are also to be defined. These could include scenarios directly identified in the FEP analysis but not according to the criteria above.

No such issues have been identified in SR-Site. There are, therefore, no residual scenarios in addition to those defined according to the procedure described in steps 2 and 3 above.

An additional task for the safety assessment is to contribute to the demonstration that best available technique, BAT, has been applied in the repository design. This requires a number of dedicated calculation cases that are not included among the selected scenarios. See further Section 2.7.

5. Combination of scenarios

For the scenario selection to be comprehensive, combinations of the scenarios and variants must be considered. This is done when all the variants and residual scenarios have been selected and analysed. The number of possible combinations could become large, even considering that mutually exclusive scenarios should not be combined, and a practical approach for handling this situation has to be adopted. The problem is further compounded by the fact that each variant may be investigated through a number of calculation cases.

Related to the issue of combination of scenarios is that of different event sequences. The sequence in which different events or aspects of the evolution occur may be important for the evolution of the repository. This is explicitly addressed within each scenario.

The evaluation of combinations of scenarios is described in Section 12.9.

2.5.9 Step 9: Analysis of selected scenarios

The analysis of the selected scenarios is divided in two steps: Analysis of containment potential and of retardation potential.

Analysis of containment potential

The containment potential is not further analysed in the main scenario; the results from the reference evolution in step 7 (Section 2.5.7) are adopted.

The additional scenarios are analysed by focussing on the factors potentially leading to situations in which the safety function in question is not maintained. In most cases, these analyses are carried out by comparison with the evolution for the main scenario, meaning that they only encompass aspects of repository evolution for which the scenario in question differs from the main scenario.

A common template is followed in the analysis of the additional scenarios. The headings in the template are the following:

- Safety function indicator(s) considered.
- Treatment of this issue in the reference evolution.
- Qualitative description of routes to this situation.
- Quantitative assessment of routes to this situation.
- Categorisation as "less probable" or "residual" scenario.
- Conclusions.

The analysis of the containment potential is carried out in Chapter 12, with the exception of scenarios related to future human actions that are analysed in Section 14.2.

Analysis of the retardation potential

This second step of the scenario analyses encompasses calculations of radionuclide release, transport and dose impacts for potential failure modes of canisters identified for each scenario in the analysis of containment potential. The purpose is to assess the retardation properties of the system for these scenarios and to quantify risk.

In general, a number of calculation cases are defined for each scenario, except for the main scenario for which consequences are encompassed by the cases analysed for the additional scenarios. The calculation cases are formulated to account for uncertainties relating to both the containment potential and the retardation potential.

Uncertainties relating to retardation potential are not included as factors in the selection and analysis of scenarios described above. Therefore, for each scenario, the safety functions related to retardation are treated in a similar manner to that in which the functions related to containment were treated in the scenario analysis described above, in order to get a comprehensive coverage of uncertainties relating to retardation. This is done through application of a similar template as for the containment potential. The number of cases this yields for each scenario is limited to a manageable number since for the most important scenarios i) the cases are similar between the scenarios, and ii) the identified failure modes in a scenario often imply that one or several barriers are impaired to the extent that they are short-circuited so yielding a reduction in the number of uncertain factors. For example, one scenario addresses the loss of containment due to corrosion. In that scenario, a risk contribution is obtained only in cases where the buffer is lost. For this situation, a calculation base case for radionuclide transport and dose is defined, and then all safety functions contributing to retardation are examined through the mentioned template to derive additional calculation cases covering uncertainties relating to retardation. The fact that the buffer is lost in this scenario reduces the number of calculation cases considerably.

The calculated risks from the main scenario and from the additional scenarios that are categorised as "less probable" (but not "residual") are summed to obtain a total risk over time for the repository.

Sensitivity analyses of the output of the probabilistic calculations to uncertainties in input data are carried out mainly by i) calculation of standardised rank regression coefficients for the evaluation of probabilistic results and ii) the formulation of a tailored regression model, based on the mathematical model used in the risk calculation.

The central results of the consequence calculations are presented in Chapter 13 whereas details are provided in the SR-Site **Radionuclide transport report**.

2.5.10 Step 10: Additional analyses and supporting arguments

In this step, a number of additional analyses, required to complete the safety assessment, are carried out. These comprise:

- The selection and analysis of FHA scenarios, including a brief description of the methodology for the selection.
- Analyses required to demonstrate optimisation and use of best available technique.
- Verification that FEPs omitted in earlier parts of the assessment are negligible in light of the completed scenario and risk analysis.
- A brief account of the time period beyond one million years.
- Natural analogues.

See further Chapter 14.

2.5.11 Step 11: Conclusions

This step includes integration of the results from the various scenario analyses, development of conclusions regarding safety in relation to regulatory criteria and feedback concerning repository design, detailed site investigations and SKB's RD&D Programme.

The discussion of compliance with the regulatory risk limit is a central part of the conclusions. This is associated with a confidence statement, discussing the confidence in the various aspects of the assessment on which the risk calculations are built.

This step also contains conclusions and feedback regarding the design of the engineered barriers and the repository. Specifically, a set of design basis cases is presented, based on the risk contributing scenarios, in agreement with applicable regulations. These take the form of specifications of loads that the repository system should withstand over time. These updated design basis cases, together with other findings from SR-Site, are used to assess the need for updating the design premises related to long-term safety /SKB 2009a/ used for developing the current design for long term safety. In addition to the design basis cases and other input to revision of the design premises, feedback is given regarding a number of detailed aspects of the design. Conclusions are also drawn regarding best available technique, BAT. This is, however, a many-faceted issue for which only part of the basis for conclusions comes from the safety assessment and in particular from the calculated risks.

See further Chapter 15.

2.5.12 Report hierarchy in the SR-Site project

As indicated in the previous section, several of the steps carried out in the SR-Site assessment result in specific reports that are of central importance for the conclusions and analyses in this main report. Table 2-1 lists these *main references* and defines the abbreviations by which they are identified in the text hereinafter. The report with the full title "FEP report for the safety assessment SR-Site" is e.g. referred to as the **FEP report**. There are also about 80 *additional references*, treating narrower issues, and that support either the main report or one of the main references. The report hierarchy is illustrated in Figure 2-3. All these reports can be downloaded from www.skb.se.

Furthermore, as mentioned in Section 1.6, one of the most fundamental input documents to the SR-Site project is the site descriptive model of the Forsmark site /SKB 2008a/. This report is briefly identified as **Site description Forsmark**. (In other contexts, the report is in brief referred to as SDM-Site Forsmark.)

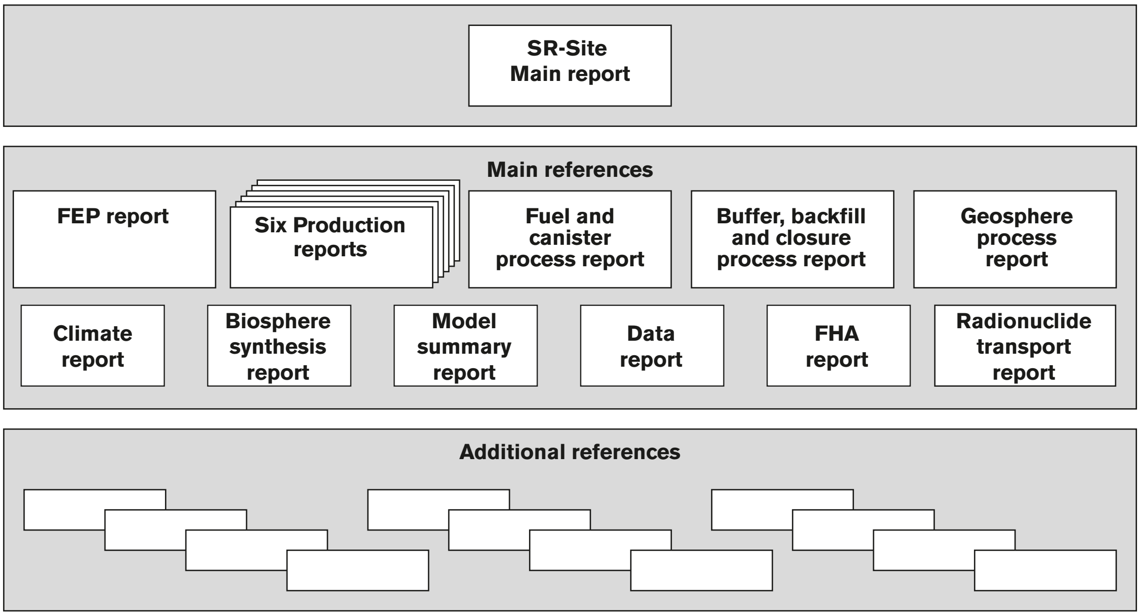


***Figure 2-3.** The hierarchy of the main and additional references to the SR-Site project. The main references support the main report. The additional references may either support the main report directly or one of the main references. The six **Production reports** include the **Spent fuel report** and the **Underground openings construction report**.*

Table 2-1. Main references in the SR-Site project. All these reports are available at www.skb.se.

Full title	Abbreviation used when referenced in this main report	Text in reference list (Chapter 16)
FEP report for the safety assessment SR-Site	**FEP report**	**FEP report, 2010.** FEP report for the safety assessment SR-Site. SKB TR-10-45, Svensk Kärnbränslehantering AB.
Spent nuclear fuel for disposal in the KBS-3 repository	**Spent fuel report**	**Spent fuel report, 2010.** Spent nuclear fuel for disposal in the KBS-3 repository. SKB TR-10-13, Svensk Kärnbränslehantering AB.
Design, production and initial state of the canister	**Canister production report**	**Canister production report, 2010.** Design, production and initial state of the canister. SKB TR-10-14, Svensk Kärnbränslehantering AB.
Design, production and initial state of the buffer	**Buffer production report**	**Buffer production report, 2010.** Design, production and initial state of the buffer. SKB TR-10-15, Svensk Kärnbränslehantering AB.
Design, production and initial state of the backfill and plug in deposition tunnels	**Backfill production report**	**Backfill production report, 2010.** Design, production and initial state of the backfill and plug in deposition tunnels. SKB TR-10-16, Svensk Kärnbränslehantering AB.
Design, production and initial state of the closure	**Closure production report**	**Closure production report, 2010.** Design, production and initial state of the closure. SKB TR-10-17, Svensk Kärnbränslehantering AB.
Design, construction and initial state of the underground openings	**Underground openings construction report**	**Underground openings construction report, 2010.** Design, construction and initial state of the underground openings. SKB TR-10-18, Svensk Kärnbränslehantering AB.
Fuel and canister process report for the safety assessment SR-Site	**Fuel and canister process report**	**Fuel and canister process report, 2010**. Fuel and canister process report for the safety assessment SR-Site. SKB TR-10-46, Svensk Kärnbränslehantering AB.
Buffer backfill and closure process report for the safety assessment SR-Site	**Buffer, backfill and closure process report**	**Buffer, backfill and closure process report, 2010**. Buffer, backfill and closure process report for the safety assessment SR-Site. SKB TR-10-47, Svensk Kärnbränslehantering AB.
Geosphere process report for the safety assessment SR-Site	**Geosphere process report**	**Geosphere process report, 2010**. Geosphere process report for the safety assessment SR-Site. SKB TR-10-48, Svensk Kärnbränslehantering AB.
Climate and climate related issues for the safety assessment SR-Site	**Climate report**	**Climate report, 2010**. Climate and climate related issues for the safety assessment SR-Site. SKB TR-10-49, Svensk Kärnbränslehantering AB.
Radionuclide transport report for the safety assessment SR-Site	**Radionuclide transport report**	**Radionuclide transport report, 2010.** Radionuclide transport report for the safety assessment SR-Site. SKB TR-10-50, Svensk Kärnbränslehantering AB.
Model summary report for the safety assessment SR-Site	**Model summary report**	**Model summary report, 2010**. Model summary report for the safety assessment SR-Site. SKB TR-10-51, Svensk Kärnbränslehantering AB.
Data report for the safety assessment SR-Site	**Data report**	**Data report, 2010**. Data report for the safety assessment SR-Site. SKB TR-10-52, Svensk Kärnbränslehantering AB.
Handling of future human actions in the safety assessment SR-Site	**FHA report**	**FHA report, 2010**. Handling of future human actions in the safety assessment SR-Site. SKB TR-10-53, Svensk Kärnbränslehantering AB.
Biosphere analyses for the safety assessment SR-Site – synthesis and summary of results	**Biosphere synthesis report**	**Biosphere synthesis report, 2010.** Biosphere analyses for the safety assessment SR-Site – synthesis and summary of results. SKB TR-10-09, Svensk Kärnbränslehantering AB.

2.6 Approach to risk calculations

2.6.1 Regulatory requirements and guidance

The quantitative acceptance criterion in Sweden for long-term safety of a nuclear waste repository is a limit on annual risk. SSMFS 2008:37 states the following: "A repository for spent nuclear fuel or nuclear waste shall be designed so that the annual risk of harmful effects after closure does not exceed 10^{-6} for a representative individual in the group exposed to the greatest risk." The conversion between effective dose and risk is to be carried out using ICRP's probability coefficient for cancer and hereditary effects of 0.073 per Sievert. An annual risk limit of 10^{-6} thus corresponds to an effective dose limit of about $1.4 \cdot 10^{-5}$ Sv/yr. (ICRP's coefficient has been updated and is now slightly lower, but this is not implemented in the Swedish regulations.)

SSM's General Guidance state the following: "The individual risk should be calculated as an annual average on the basis of an estimate of the lifetime risk for all relevant exposure pathways for every individual."

As already mentioned in Section 2.4, according to the guidance to SSM's Regulations, the quantitative risk criterion is applicable as a quantitative regulatory limit during approximately the first one hundred thousand years, and thereafter as one of several indicators forming a basis for discussing the protective capability of the repository.

It is furthermore important to note that scenarios relating to future human actions should not be included in the risk calculation according to SSM's General Guidance.

2.6.2 Application in SR-Site

This section describes some basic aspects of how compliance with SSM's risk criterion is demonstrated in SR-Site. Most of the material below has been presented and discussed at an NEA Workshop on the role of risk in safety assessments /NEA 2005/ and at an international conference on probabilistic safety assessment and management /Hedin 2004a/.

Scenario disaggregation

In principle, the product of dose consequences and likelihoods of all possible future evolutions of the repository should be weighed together and presented as a time-dependent risk. The spectrum of possible evolutions is, however, very wide and cannot be captured in a detailed sense. This is also recognised in SSM's Regulations and associated General Guidance.

The usual approach taken in safety assessments, and also in SR-Site, is to work with scenarios and variants that are designed to capture the broad features of a number of representative possible future evolutions. Together, these are intended to give a reasonable coverage of possible future exposure situations. In principle, conditional risks could be calculated for each scenario and variant and then weighed together using the probability for each scenario/variant. However, in practice, scenario probabilities often have to be pessimistically overestimated, see further the subsection "Overestimation of risk" below. Furthermore, each variant, represented by a specific calculation case, may be evaluated probabilistically in order to determine the mean exposure given the data uncertainties for the particular variant.

The approach of calculating risk as a sum of risk contributions from a number of scenarios constrains the way in which scenarios are selected and defined. It must be possible to logically explain the summation and the set of scenarios should be comprehensive in the sense that all relevant future evolutions are covered.

A "normal evolution" scenario with a high probability of occurrence must e.g. contain initially defective canisters and other barrier insufficiencies, if such are likely when the entire ensemble of canisters and deposition holes in the repository is considered.

Since SSM's General Guidance states that the risk criterion concerns a repository undisturbed by man, scenarios involving direct intrusion into the repository are excluded from the risk summation. Also human actions that disturb the immediate environment of the repository, e.g. the local groundwater flow field, are considered in the treatment of future human actions in Section 14.2, but excluded from the risk summation.

Overestimation of risk

The formulation of scenarios, variants and calculation cases, and the subsequent weighing together of these to give a total risk aims at an over prediction of risk. SSM's regulation requires that the annual risk should be less than 10^{-6}. There are a number of uncertainties that cannot be managed quantitatively in any other rigorous manner from the point of view of demonstrating compliance than by pessimistic assumptions.

Another situation in which risk has to be overestimated concerns scenario probabilities. Regarding e.g. future climate, both repetitions of conditions reconstructed for the past 120,000 year glacial cycle and an alternative where this development is considerably perturbed by a global warming effect can be envisaged. Although the two are mutually exclusive, both must be regarded as possible. In the risk summation, the logical position is adopted that the summed consequence of a set of mutually exclusive scenarios can, at any point in time, never exceed the maximum of the individual scenario consequences. For scenarios and variants where defensible probabilities are difficult to derive, a scenario or variant giving high consequences can pessimistically be assigned unit probability and other scenarios and variants yielding lower dose impacts can be "subsumed" under the one with the more severe consequences.

Although the primary aim with risk calculations is to demonstrate compliance, there is also the clear ambition of clarifying the sensitivities of the calculation results. For this aim, the calculation cases should be, in principle, as realistic as possible in capturing uncertainty. One quantitative tool for this is the use of probabilistic evaluations of calculation cases followed by sensitivity analyses of the results.

It is concluded that pessimistic simplifications should be avoided where a sound scientific basis exists for a quantitative treatment and further that the pessimistically neglected features of the system should be included in a discussion of sensitivities.

Size of the exposed group

The size of the group to which the above risk limit is to be applied must be defined in order to evaluate compliance with the risk criterion. No detailed definition is given in SSMFS 2008:37. In its General Guidance SSM however states the following: "One way of defining the most exposed group is to include the individuals that receive a risk in the interval from the highest risk down to a tenth of this risk. If a larger number of individuals can be considered to be included in such a group, the arithmetic average of individual risks in the group can be used for demonstrating compliance with the criterion for individual risk in the regulations. One example of such exposure situation is a release of radioactive substances into a large lake that can be used as a source of drinking water and for fishing.

If the exposed group only consists of a few individuals, the criterion of the regulations for individual risk can be considered as being complied with if the highest calculated individual risk does not exceed 10^{-5} per year. An example of a situation of this kind might be if consumption of drinking water from a drilled well is the dominant exposure path. In such a calculation example, the choice of individuals with the highest risk load should be justified by information about the spread in calculated individual risks with respect to assumed living habits and places of stay."

The detailed application of these two options in SR-Site is further developed in connection with the consequence calculations in Chapter 13.

Time frames

Risk calculations are carried out for a one million year time frame in SR-Site. In accordance with SSM's General Guidance, strict compliance with the risk limit is evaluated in a 100,000 years time frame. For longer times, also in accordance with SSM's General Guidance, the results of the risk calculation are used to discuss the protective capability of the repository and how this capability can be improved.

Time dependent risk or peak over entire assessment period?

An upper bound on the peak of the time dependent risk may be calculated by determining the peak annual effective dose[5] over the one million year assessment period in each realisation. The mean value of the determined distribution of peak doses is then compared with the effective dose criterion. While this is a correct way of putting an upper bound on risk, it is more informative and also in agreement with the SSM's Regulations to calculate the mean annual effective dose at each point in time and require that this quantity never exceeds the effective dose corresponding to the risk criterion of 10^{-6}. The two methods are sometimes referred to as "the mean of the peaks" and "the peak of the mean". The "peak of the mean" interpretation is meaningful in the sense that all exposure pathways to

[5] The annual effective dose is the sum of the annual effective dose from external exposure and the annual committed effective dose from internal exposure.

hypothetical individuals living in the future are considered whereas the "mean of the peaks" concept is more difficult to interpret. In SR-Site, risk as a function of time is presented by weighing together the time-dependent mean annual effective doses from each scenario to obtain a time-dependent risk.

Risk dilution

The term "risk dilution" is sometimes used to denote a situation where a higher degree of uncertainty in input parameters, i.e. a broader input distribution, leads to a lower mean value of an output quantity e.g. mean dose or risk /NEA 1997b/. A seemingly paradoxical situation arises where less knowledge implies a safer repository if the mean value to a highly exposed individual at a certain point in time is used as the safety indicator. Less knowledge will spread the dose over more individuals and over longer times. The total exposure to all individuals over all times could be the same or larger, whereas more precise knowledge will "concentrate" the risk to fewer individuals and shorter periods of time. This can e.g. be the case when there is uncertainty concerning the time of an event that would lead to canister rupture. The dose consequence for a given time could then depend strongly on the assumed time at which the rupture occurred. Averaging over alternative situations in which canister rupture and thus peak dose occurs at different points in time would reduce the resulting mean value at any point in time and more so the larger the span of possible rupture times.

This effect is inherent in the concept of risk as defined in SSM's regulation and is thus an inevitable consequence of a risk criterion which is to be applied as a function of time and where the quantity to be determined is the mean value considering all relevant uncertainties. Therefore, SSM's General Guidance also requires that the issue of risk dilution is addressed when the consequences of releases from the repository are assessed.

A related phenomenon concerns biosphere development during the expected long periods of permafrost or glacial conditions. To illustrate, it is assumed that appreciable doses to man could occur only during temperate periods, and that these periods, as suggested by historical evidence relevant to Sweden, in the long run will prevail in total during about ten percent of the time, but that the temporal location of these temperate intervals cannot be predicted beyond, say 10,000 years into the future. In principle, this situation could be handled by simulating a number of future situations where the onsets of the temperate periods are allowed to vary randomly beyond 10,000 years. Averaging over all these results would, at each point in time beyond 10,000 years, yield a dose consequence a factor of ten smaller than that obtained during a temperate climate period. This simplistic example demonstrates another type of risk dilution, again caused by an uncertainty in the point in time of the occurrence of a phenomenon, which could in principle be compatible with the Swedish risk criterion. The effect will however be avoided in the safety assessment e.g. by assuming the same temporal sequence of climate types in each simulation or by assuming today's biosphere.

For SR-Site several conclusions are drawn from the above, as set out below.

- A broader input data distribution is not necessarily pessimistic, not even if it is broadened towards the high consequence end. Thus, care must be taken in assigning input data distributions so that input data distributions that might influence the calculation end-point in this way are not unduly broadened.
- Disaggregated calculations and disaggregated discussions of the results of more integrated calculations are necessary from the point of view of capturing risk dilution and such calculations and discussions are, therefore, included in SR-Site. A simple but effective means of avoiding risk dilution when its cause has been identified is to illustrate the effect by replacing probabilistic input data of e.g. canister rupture times with a fixed time. This is one approach taken in SR-Site.
- For certain situations with short term releases, a method suggested in Appendix 1 to SSMFS 2008:37 is used to address risk dilution.
- Another option for capturing risk dilution effects is to complement a "peak of the mean" calculation with a "mean of the peaks" calculation, as described in the previous section. This is, however, used only for illustrative purposes in SR-Site since the result of such a calculation is difficult to interpret, as discussed above.

2.6.3 Alternative safety indicators

The dose/risk safety indicator provides a measure of radiological impact on future humans due to the existence of the repository. Several aspects of biosphere development are highly uncertain, even over a relatively short time perspective. The evaluation of safety depends on a number of assumptions made in order to handle these uncertainties. It is, therefore, of interest to complement both the dose/risk indicator and the evaluation of impacts on non-human biota with alternative indicators that do not require detailed assumptions about the biosphere or concerning human habits.

The recommendations accompanying SSMFS 2008:21 mention that, for distant futures, the dose indicator can be complemented with other safety indicators, e.g. concentrations in groundwaters or near-surface waters of radionuclides from the repository or the calculated fluxes of radionuclides to the biosphere.

A problem with alternative indicators is that there is, in general, no obvious criterion with which the calculated quantities can be compared. In some cases, calculation results can be compared with natural concentrations or fluxes at the site or elsewhere. However, such criteria do not provide points of reference for man-made radionuclides. The problem can be partly overcome by comparing naturally occurring sum concentrations/fluxes of α- and β-emitters to the corresponding repository related quantities, or by comparing overall toxicities by scaling by dose per unit intake values.

It is also noted that the safety function indicators (Chapter 8) are a type of alternative safety indicator related to barrier conditions, rather than to concentrations and fluxes, with the indicator criteria being reference values with which to compare the outcome of an analysis of the condition in question. However, whereas the safety function indicators relate to the functioning of sub-systems, the safety indicators provide a measure of the overall system performance.

EU SPIN Project

An EU project /Becker et al. 2002/ concludes that two alternative indicators could preferably be used to complement the dose indicator. These are:

- Radiotoxicity concentration in biosphere water: preference for medium time frames, i.e. several thousand to several tens of thousands of years.
- Radiotoxicity flux from the geosphere: preference for later time frames.

The project also reports on reference values that could tentatively be used for comparisons to calculated concentrations and fluxes of radionuclides from the repository. Regarding radiotoxicity flux from the geosphere, an indicative reference value of 60 Sv/y for a typical area of 200 km^2 was suggested.

Finnish activity release constraints

The Finnish Radiation and Nuclear Safety Authority STUK has issued activity release constraints to the environment /STUK 2001/.

These nuclide specific constraints are defined for long-lived radionuclides only. The effects of their short-lived progeny have been taken into consideration in the constraints defined for the long-lived parents. The nuclide-specific release rate constraints are:

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

The constraints apply to activity releases that arise from the expected evolution scenarios and that may enter the environment after several thousands of years, whereas dose rate constraints are applied in the shorter term. In applying the above constraints, the activity releases can be averaged over 1,000 years at the most. The sum of the ratios between the nuclide-specific activity releases and the respective constraints shall be less than one. It should be noted that the Finnish regulator has derived these constraints partly based on a set of reference biospheres considered possible in the future at the planned disposal site, Olkiluoto at the coast of the Baltic Sea, and partly on natural fluxes of radionuclides established for similar environments. The reference values of the Finnish regulatory guide are thus not directly applicable for other disposal concepts and sites /Becker et al. 2002/. However, both the disposal concept and the sites considered in Sweden are similar to those for which the Finnish activity release constraints have been developed.

Other studies

An SKI/SSI study /Miller et al. 2002/ compiled from the published literature a substantial database of elemental abundances in natural materials and, using these data, calculated a range of elemental and activity fluxes arising due to different processes at different spatial scales. The authors conclude that these fluxes should be comparable to results from safety assessment calculations.

IAEA has published a study entitled "Safety Indicators in Different Time Frames for the Safety Assessment of Underground Radioactive Waste Repositories" /IAEA 1994/ and is currently conducting a research programme on natural concentrations and fluxes.

Implications for SR-Site

Four alternative indicators to risk are used in SR-Site; release of activity from the geosphere, radiotoxicity flux from the geosphere, concentrations of radionuclides in ecosystems and fluxes of radionuclides. The following reference values are used when evaluating these indicators:

- The Finnish activity constraints. These constraints are strictly applicable only in the Finnish regulatory context, but nevertheless are deemed useful as reference values for SR-Site.
- The reference value for radiotoxicity flux from the geosphere suggested by the SPIN project.
- Measured concentrations of naturally occurring radionuclides in ecosystems at the Forsmark site or other, comparable sites.
- Naturally occurring fluxes of radionuclides at the site, in particular of U-238 and Ra-226.

2.7 BAT and optimisation

2.7.1 Introduction

According to Swedish legislation, a licence application for a final repository needs to address the issues of best available technique (BAT) and optimisation. The general account of BAT is given in a dedicated Annex to the licence application (in Swedish). The licence application is also supported by a report providing an account of i) alternative methods and the reasons for choosing the KBS-3 method and ii) the technical development of the KBS-3 method with motivations as to materials selected, dimensions of barriers etc.

To comply with the detailed regulations regarding long-term safety issued by SSM, some aspects of the demonstration of BAT need to be addressed in the assessment of long-term safety supporting the licence application, i.e. in the SR-Site reporting. The following is a more detailed account of these requirements and of the approach to addressing them.

2.7.2 Regulatory requirements

Regarding optimisation and best available technique, the General Guidance to SSMFS 2008:37 states the following:

"The regulations require that optimisation must be performed and that best available technique should be taken into account. Optimisation and best available technique should be applied in parallel with a view to improving the protective capability of the repository.

Measures for optimisation of a repository should be evaluated on the basis of calculated risks.

Application of best available technique in connection with final disposal means that the siting, design, construction, operation and closure of the repository and appurtenant system components should be carried out so as to prevent, limit and delay releases from both engineered and geological barriers as far as is reasonably possible. When striking balances between different measures, an overall assessment should be made of their impact on the protective capability of the repository.

In cases where considerable uncertainty is attached to the calculated risks, for instance, in analyses of the repository a long time after closure, or analyses made at an early stage of the development work with the repository system, greater weight should be placed on best available technique.

In the event of any conflicts between application of optimisation and best available technique, priority should be given to best available technique.

Experiences from recurrent risk analyses and the successive development work with the repository should be used in the application of optimisation and best available technique."

SSMFS 2008:21 states that the "barrier system shall be designed and constructed taking into account the best available technique" and refers to the Swedish Environmental Code. In the General Recommendations related to SSMFS 2008:21, it is stated that "The use of the best available technique means that the technology, from a technical and economic standpoint, shall be industrially feasible for application within this area. This means that the technique must be available and not merely at the experimental stage. However, the technique does not have to be available in Sweden (see bill 1997/98:45, Part I, p 215 ff for details)".

2.7.3 General issues regarding optimisation and best available technique

A general account of the use of best available technique (BAT) is a broad issue spanning from the selection of method for the management of nuclear waste to fine details of the selected method. Only a limited part of this broad issue can and should be addressed in the safety assessment of the preferred method. Here, the account of BAT is, therefore, confined to the KBS-3 method with vertical deposition, using copper/cast iron canisters, buffer and backfill at the selected site. Safety relevant aspects of the site selection are discussed in a separate report supporting the license application. As was demonstrated in the SR-Can assessment, Section 13.3.4, it is, to some extent, possible to discuss e.g. the choice of materials and dimensions from the perspective of BAT, based on the results of the safety assessment. This role for the safety assessment regarding its contribution to the discussion of optimisation and BAT is also in agreement with the view expressed in SKI's and SSI's joint review of SKB's SR 97 assessment, Section 3.3.6 of /SKI/SSI 2001/. The broader issue of BAT is discussed in a dedicated document supporting SKB's application for a final repository, see above.

The issue of BAT is also closely related to that of feedback to repository design, as also stated in the General Guidance to SSMFS 2008:37. As also acknowledged in the regulations, the development of a repository system is carried out in steps, with safety evaluations at appropriate points of the development. This means that the developer is not in a position to finally claim optimisation and use of BAT until much of the iterative development work has been finalised. At earlier stages, the account of optimisation and BAT is rather a framework for discussing feedback to remaining development needs. The discussion conducted in relation to the SR-Can assessment is to a considerable extent an example of the latter role for optimisation and BAT.

2.7.4 Optimisation vs BAT

In the General Guidance to SSMFS 2008:37, optimisation is emphasised more for the initial period after closure. It is also stated that optimisation should be carried out with respect to calculated risks. As evidenced by the account of the calculated risk in the SR-Can assessment (Section 13.3.2 /SKB 2006a/), the assessed risk is negligible for tens of thousands of years into the future, suggesting that optimisation is of limited relevance during this initial period that is emphasised in the regulations. However, as also mentioned in the regulations, the considerations of optimisation and BAT should be applied in parallel. In fact, it is often difficult to clearly distinguish the two. The discussion of BAT, when based on results from the safety assessment of the preferred method, is frequently 'reduced' to an account of optimisation of the selected solution, since there are in general no alternative techniques to choose between or analyse in the safety assessment. (The term "optimisation" is used in the same sense as in SSMFS 2008:37, i.e. "keeping the radiation doses to mankind as low as reasonably achievable, economic and social factors taken into account", the so called ALARA principle.)

In the General Guidance to SSMFS 2008:37, it is noted that, for the time period between 100,000 years and one million years, the results of the risk calculation should be used to discuss measures to improve the protective capability of the repository if the risk limit is exceeded.

2.7.5 Conclusions relating to methodology for the SR-Site assessment

Based on the above, it is concluded that the calculated risk results in the assessment of long-term safety need to be evaluated from the point of view of BAT.

To achieve this, analyses of the sensitivity of the risk with respect to important barrier dimensions, layout rules, repository depth etc. are carried out as part of the additional analyses following the risk calculation, see further Section 14.3.

Based on the results of the sensitivity analyses, a discussion of the compliance with this aspect of BAT is also provided, in Section 14.3.

2.8 Overall information/uncertainty management

A safety assessment handles a vast amount of information of qualitative and quantitative nature, including the uncertainties associated with that information. This section gives an overview of issues related to information and uncertainty management in SR-Site. Since this issue permeates the entire analysis, the overview is, in part, a summary of the different steps of the methodology described in Section 2.5, but with emphasis on information/uncertainty management. In all management of uncertainty, it is important to consider the significance of the uncertain issue relative to the purposes of the safety assessment.

As a background, Section 2.8.1 gives a brief description of the different types of uncertainty that have to be managed in the safety assessment.

2.8.1 Classification of uncertainties

There is no unique way in which to classify uncertainties in a safety assessment. The classification adopted below is, however, compatible with international practice /NEA 1991, 1997a/ in this type of analysis. SKB has previously discussed the classification and nature of uncertainties in detail, see e.g. /SKB 1995, Section 3.4/ and /Andersson 1999, Section 2.1/ and the SR-Can main report /SKB 2006a, Section 2.7/ of which the following is an update. Here, only a brief outline is given, setting the context for the presentation of the management plan.

The safety assessment is built on the analysis of how a system with an initial state evolves as a result of actions on the system by a number of internal processes and external influences/events. From this description, a number of issues regarding uncertainties can be identified, as listed below.

- How well is the initial state known, qualitatively and quantitatively, i.e. are all important aspects of the initial state identified and how well can they be quantitatively described?

- Have all relevant internal processes been identified in the relevant time frames? How well are the process mechanisms understood?
- Have all relevant external events and phenomena been identified? How well can they be quantified?
- How can a representative account of the system evolution be given, taking into account all the types of uncertain factors mentioned above? How well can the internal processes be represented mathematically to give a realistic account of the system evolution? How well are all the input data necessary for the quantification of the system evolution known?

In defining a structure for a rigorous approach to the above issues, it is customary /NEA 1997a/ to describe uncertainty in the categories; system/scenario uncertainty, conceptual uncertainty and data uncertainty. A general conclusion from international collaboration efforts in the area of assessment methodology is that there is no unique or correct way to describe or classify uncertainty. Rather, in any safety assessment, it is important to make clear definitions of the use of different terms in this area, in the light of the results from international efforts such as the compilation of lessons learnt from ten performance assessment studies /NEA 1997a/.

In SR-Site, the following broad definitions are used.

System uncertainty concerns comprehensiveness issues, i.e. the question of whether all aspects important for the safety evaluation have been identified and whether the analysis is capturing the identified aspects in a qualitatively correct way, e.g. through the selection of an appropriate set of scenarios. In short, have all factors, FEPs, been identified and included in a satisfactory manner or has their exclusion been appropriately justified?

Conceptual uncertainty essentially relates to the understanding of the nature of processes involved in repository evolution. This concerns not only the mechanistic understanding of a process or set of coupled processes, but also how well they are represented, and what is not represented, in a possibly considerably simplified mathematical model of repository evolution.

Data uncertainty concerns all quantitative input data used in the assessment. There are a number of aspects to take into account in the management of data uncertainty. These include correlations between data, the distinction between uncertainty due to lack of knowledge (epistemic uncertainty) and due to natural variability (aleatoric uncertainty) and situations where conceptual uncertainty is treated through a widened range of input data. The input data required by a particular model is in part a consequence of the conceptualisation of the modelled process, meaning that conceptual uncertainty and data uncertainty are to some extent intertwined. Also, there are several conceivable strategies for deriving input data. One possibility is to strive for pessimistic data in order to obtain an upper bound on consequences in compliance calculations. Another option is the full implementation of a probabilistic assessment requiring input data in the form of probability distributions. These aspects are further discussed in Chapter 9 and in the **Data report**.

The plan presented in Section 2.8.3 demonstrates how all the discussed types of uncertainty are managed in the safety assessment.

2.8.2 Need for stylised examples

The management of uncertainties is most comprehensive for the behaviour of the inner parts of the system, which provide the containment and retardation safety functions. The biosphere and the external conditions are handled in a more stylised manner, i.e. through simplified representations where the important aspects of these sub-systems are captured, often in a pessimistic fashion. These latter parts do not incorporate principal safety related features of the system and they are too complex to be modelled in detail in the safety assessment.

The local biosphere is by definition a part of the system, i.e. it lies within the system boundaries and biosphere uncertainties should thus be managed in the same way as for other internal parts. However, in the biosphere, the list of processes determining the system development is long and the system in which they occur is highly inhomogeneous, including a number of different ecosystems each with a large number of components. Furthermore, the time scale on which the biosphere changes is in general considerably shorter than for other parts of the system, and the interactions with humans are stronger and associated with partly irreducible, large uncertainties. Although some aspects of the development of the biosphere at a particular location can be reasonably forecast in maybe a 1,000 year perspective, a large part of the description, particularly of human behaviour has to be through stylised examples, see further Section 13.2.

Also in relation to the effects of external conditions, uncertainty management largely has to be through stylised examples devised to cover the range of possible future evolutions, e.g. regarding climate change. A detailed treatment of all the processes involved in climatic development is outside the scope of the safety assessment. Climate research is furthermore a rapidly evolving field of science, where uncertainties are fundamental and in part irreducible. The approach is instead to follow the development of the field, and from an exemplified future reference glacial cycle derive a number of complementary stylised possible evolutions that together bound the possibilities of what could be expected in the future. In particular, extreme conditions that could have a negative effect on repository safety are captured in these examples. These conditions include:

- Maximum glacial overburden and the resulting hydraulic/mechanical pressures and hydraulic/mechanical loads on the bedrock.
- Intrusion of waters of extreme composition, such as oxygenated glacial melt water of low ionic strength.
- Extreme surface boundary conditions for groundwater flow possibly leading to high groundwater fluxes at repository level or groundwater movements that could cause intrusion of deeply lying saline groundwaters.
- Conditions leading to extreme permafrost depths.

2.8.3 Uncertainty management; general

The purpose of the safety assessment affects the management of uncertainties. In this context, the purpose of the assessment is essentially two-fold:

- To assess compliance with Swedish regulations.
- To give feedback to design, research and development and further detailed site investigations as the repository is constructed.

The first purpose can, if there are sufficient safety margins, be largely accomplished by a pessimistic handling of many uncertainties. The second, however, requires more sophisticated management in order to determine quantitatively which uncertain factors and open design issues affect safety most.

In the following, the broad features of the management of uncertainties in SR-Site are outlined. QA aspects of the handling are further discussed in Section 2.9.

System uncertainty

System uncertainty is generally handled through the proper management of FEPs in the FEP database according to the routines described in the **FEP report** and summarised in Chapter 3 of this report.

The database structure and FEP management routines have been set up to assure that the following information is obtained.

- A sufficient set of initial conditions. This is obtained by including all initial state FEPs in the database. These are, however, often formulated in general terms and have to be expressed in a way that is specific to the KBS-3 system. This is done through the systematic documentation of a reference initial state in accordance with the description in the **Production reports** and by using that initial state as a starting point for alternative initial states, where relevant.
- A sufficient set of internal, coupled processes. This is obtained by including in the assessment all relevant process FEPs in the database. It is important to note that the database already from the start includes the result of several earlier exercises aiming at process identification for the KBS-3 concept. This is further described in Section 7.1.1. Influences between processes are handled, in the **Process reports**, by systematically going through a set of defined physical variables

that could mediate influences and by the systematic treatment of boundary conditions for each process. These procedures are further described in Section 7.3. Hence, in addition to including FEPs describing influences and couplings, the procedures for process documentation are set up in a way that enforces a systematic search for such influences.

- A sufficient set of external influences. This is obtained by including in the assessment all relevant external FEPs and by structuring the documentation of these in the **Climate report** in a format similar to that used for the internal processes, see further Chapter 6.

Scenario selection

Another aspect of system uncertainty concerns the selection of a sufficient set of scenarios, through which all relevant FEPs are considered in an appropriate way in the analysis. The selection of scenarios is a task of a subjective nature, meaning that it is difficult to propose a method that would guarantee the correct handling of all details of scenario selection. However, several measures have been taken to build confidence in the selected set of scenarios:

- A structured and logical approach to the scenario selection, see further Chapter 11.
- The use of safety function indicators in order to focus the selection on safety relevant issues, see Chapter 8.
- The use of bounding calculation cases to explore the robustness of the system to the effects of alternative ways of selecting scenarios, including unrealistic scenarios that can put an upper bound on possible consequences.
- QA measures to ensure that all FEPs have been properly handled in the assessment.
- The use of external reviews.

Conceptual uncertainty

The handling of conceptual uncertainty for internal processes is essentially described in the **Process reports**. For each process, the knowledge base, including remaining uncertainties, is described and, based on that information, a handling of the process in the safety assessment is established. (Uncertainty regarding influences between processes can be seen as either system uncertainty or conceptual uncertainty; it is described as system uncertainty above.)

Through the use of a defined format for all process descriptions, see Section 7.3, it is assured that the processes and their associated conceptual uncertainties are described in a consistent manner. External reviews of central parts of the process documentation have also been performed.

Conceptual uncertainty for external influences is handled in a more stylised manner, essentially through the definition of a sufficient set of scenarios and by using state-of-the-art models for the quantification of external influences, e.g. ice models for the modelling of glacial cycles. Another method is the use of bounding cases that ensure that the consequences are overestimated.

Data uncertainty

Data uncertainties are handled according to the routines described in Chapter 9 and further in the **Data report**.

Quality assurance is obtained through the use of a template for data uncertainty documentation, through clearly defined roles for participating experts and generalists and by the use of external reviews prior to finally establishing input data for the assessment.

Modelling

An essential part of the assessment concerns the quantification of both repository evolution and dose and risk consequences through mathematical modelling. Apart from requiring appropriately defined models that represent relevant conceptualisations of the processes to be modelled and quality assured input data, this step requires:

- Good model documentation, including results of code verification and results of benchmarking against other models.
- Procedures to detect and protect against human error in the execution of the models.

A dedicated **Model summary report**, compiled according to a pre-established template describes models used in the assessment and provides references to more detailed descriptions of the models, including quality assurance aspects. The mapping of processes to models, see Chapter 7, provides an overview of the models used. A guiding principle is that models and data should be documented in sufficient detail to allow calculations to be reproduced and audited.

Human errors can be minimised e.g. by formal procedures for checking that input data are correct and by the use of alternative, often simplified, models for crucial aspects of quantification. An example of the latter is given in calculations of radionuclide transport and dose in Chapter 13.

2.8.4 Integrated handling of uncertainties

As mentioned above, uncertainties can be broadly categorised as system uncertainty, conceptual/model uncertainty and data uncertainty. It needs to be assured that all these uncertainties are appropriately handled in the assessment. Since the distinction between the main scenario and additional scenarios is largely related to uncertainties and the likelihood of occurrence of relevant phenomena, the overall strategy for handling of uncertainties is closely related to the selection of scenarios.

The general discussion on handling of uncertainties is further developed below, based on the process of scenario selection.

A first step of integration: The reference evolution

The analysis of the reference evolution is divided into a number of sub-analyses covering different time frames and various, essentially thermal, mechanical, hydraulic and chemical, aspects of the evolution within a particular time frame.

Many of the analyses in the reference evolution aim at demonstrating that a certain criterion is met, i.e. they are bounding analyses, establishing favourable conditions for subsequent parts of the assessment. The results of these analyses essentially facilitate subsequent parts of the assessment and do not, in general, require any further evaluation of uncertainties. For example, the peak buffer temperature is calculated in a bounding fashion and demonstrated to lie below the safety function indicator of 100°C with all uncertainties considered, thus allowing the exclusion of mineral transformations in the buffer.

Other analyses lead to results that need to be considered in a more profound and often quantitative fashion in subsequent analyses. For examples, the result of the hydrogeological analyses point to both conceptual and data uncertainties that need to be considered in, e.g. radionuclide transport calculations.

Thus, in terms of uncertainties, the analyses in the reference evolution aim at reducing the number of uncertainties requiring further consideration and at identifying and quantifying uncertainties that need to be propagated to subsequent parts of the assessment.

To obtain a systematic handling of uncertainties in the reference evolution, each sub-analysis is concluded with a reporting of uncertainties in the results. The need for propagation of any uncertainties to subsequent parts of the safety assessment is also reported.

After completion of the analysis of the reference evolution, an account of the identified uncertainties is given in table format for subsequent use in the selection of scenarios and calculation cases for consequence analysis of the scenarios.

The main scenario

As mentioned in Section 2.5.8, the main scenario is based on the reference evolution analysed in Chapter 10 and thus covers a reference initial state, based on a realistic description of the repository system immediately after closure and a realistic description of the site with uncertainties, based on the results of site investigations and subsequent site descriptive modelling. Furthermore, all processes identified as relevant for long-term safety are addressed in the main scenario, in accordance with documentation in the **Process reports**. The main scenario also covers reference external conditions, essentially a repetition of the last glacial cycle and an alternative case with an extended period of temperate climate assumed to be due to an increased greenhouse effect. Future human actions, like drilling or other use of or influences on the host rock are, by definition, not included in the main scenario.

The following is a brief account of how the three classes of uncertainty defined in Section 2.8.1 are addressed in the reference evolution and hence in the main scenario.

General evolution

System uncertainty, FEPs included: All identified internal processes are considered and included in accordance with prescriptions developed in the **Process reports** and summarised in Table 7-2 through Table 7-6. External conditions are included as a base case in the form of a model reconstruction of the last glacial cycle, the Weichselian, and in a more stylised global warming variant of the main scenario. This means that *all external, climate related processes/phenomena are considered* in the main scenario. The extent to which the processes and phenomena are addressed is, however, limited to the range covered by the two variants, meaning that e.g. more extreme glacial loads than those in the reference evolution are not considered. FEPs related to future human actions, FHA, are by definition excluded from the main scenario, as are FEPs related to altered initial states like an abandoned repository that has not been completely backfilled and sealed.

Conceptual uncertainty: Models are selected, taking into consideration conceptual uncertainty, according to prescriptions in the **Process reports** (summarised in Table 7-2 through Table 7-6) and the **Climate report**. In many cases, conceptual uncertainties are handled pessimistically. They may also be addressed by formulating variant representations, as in the case for groundwater flow models. These models constitute the reference conceptual models for the repository evolution in SR-Site. They are summarised in the **Model summary report**.

Data uncertainty: Input data for modelling, with uncertainties, are taken from the **Data report** for many of the calculations related to the general evolution in the main scenario. Reference initial state data, which can imply a range of conditions, are used in accordance with the definition of the main scenario.

Radiological consequences

All situations where radiological consequences occur are treated more exhaustively in the additional scenarios. Radiological consequences of the main scenario, including uncertainties, are subsumed under relevant cases covering the additional scenarios, see below.

Additional scenarios based on potential loss of safety functions

A main purpose of the selection and analysis of a number of additional scenarios based on the potential loss of safety functions is to evaluate the effect of uncertainties not covered in the main scenario and, if appropriate, to include additional scenarios in the overall risk summation if this additional uncertainty analysis so requires.

In each of these additional scenarios, the handling of relevant aspects in the main scenario is revisited and the handling of relevant system, conceptual and data uncertainties is extended or modified, as appropriate.

General evolution

System uncertainty, FEPs included: All relevant FEPs potentially affecting the safety function under consideration are included, based on the FEP chart and the SR-Site FEP catalogue.

Conceptual uncertainty: Conceptual uncertainty beyond that covered by the reference conceptual models used in the main scenario is considered.

Data uncertainty: Input data beyond those used in the main scenario are considered, e.g. initial state conditions beyond the reference initial state and external conditions beyond those covered by the two variants of the main scenario.

As only those aspects of the general evolution relevant for the safety function(s) under consideration are considered, the analysis of the general evolution is much less extensive than that in the main scenario.

Radiological consequences

System uncertainty, FEPs included: All FEPs considered in the general evolution are included indirectly, since data for the consequence calculations are derived from the result of the analysis of the system evolution. All FEPs directly related to radionuclide transport in the engineered barriers and in the geosphere are included in accordance with prescriptions in the **Process reports** (summarised in Table 7-2 through Table 7-6). The biosphere models are discussed in /Anderssson 2010, Löfgren 2010, Aquilonius 2010, Lindborg 2010/.

Conceptual uncertainty: Models for radionuclide transport are selected, taking into consideration conceptual uncertainty, according to prescriptions in the **Process reports** (summarised in Table 7-2 through Table 7-6). In many cases, conceptual uncertainties are handled pessimistically. They may also be addressed by formulating variant representations, as is the case for groundwater flow models. These models constitute the reference conceptual models for the radiological consequence calculations in SR-Site. They are summarised in the **Model summary report**.

Conceptual uncertainties for the biosphere models are discussed in /Avila et al. 2010/ and the **Biosphere synthesis report**.

Data uncertainty: All data for the consequence calculations are given in the **Data report**. In most cases, input data uncertainties, frequently also those originating from conceptual uncertainties, are quantified in the form of a set of input data distributions. Correlations between these distributions are considered and included as appropriate. Uncertainty of input data to the biosphere modelling is considered in /Nordén et al. 2010, Anderssson 2010, Löfgren 2010, Aquilonius 2010, Lindborg 2010/.

If the altered evolution does not imply canister failures, then failures related to the safety function under consideration are postulated and the scenario is classified as a residual scenario. The consequence calculations for such cases often provide bounding estimates e.g. in a case where all canisters are assumed to fail at a specific time.

FHA scenarios

FHA scenarios obviously include FHA FEPs that are, by definition, not included in the main scenario. The aspects of the general evolution that are not affected by the FHA FEPs are assumed to develop as in the main scenario, meaning that conceptual and data uncertainty for these aspects are handled as in the main scenario. Aspects related to the FHA FEPs are handled in a stylised manner that gives a reasonable coverage of uncertainties. The case of an open, abandoned repository is treated similarly.

2.8.5 Formal expert elicitations

A formal expert elicitation is a tool for assessing uncertainty in input to a safety assessment. In the SR-Can assessment, formal expert elicitations were not utilised, with the exception of a study carried out by SSI regarding the likelihood of large earthquakes occurring in the vicinity of the repository /Hora and Jensen 2005/. The result of that limited study was used as one of several inputs to determine this likelihood in SR-Can.

In planning SR-Site, an evaluation of candidate issues for expert elicitations was made and the following criteria were used to determine whether an issue could be considered for an elicitation:

- The issue should be associated with large uncertainties that have a considerable impact on the assessed level of safety.
- A formal expert elicitation can be deemed to contribute to the reduction of these uncertainties in addition to what is achievable through other means established in the methodology for the assessment (evaluation of conceptual uncertainties in **Process reports**, of data uncertainty in the **Data report**, through quality assured modelling, etc. all of these leading to a well motivated and often pessimistic handling of the issue in the assessment).

The conclusion of the evaluation was that, although a number of uncertainties could in principle be amenable to a formal expert elicitation, no issue was identified for which both the above criteria apply.

The procedures established for the qualification of processes and data in SR-Site, the considerable and concerted research activities on critical issues, the comprehensive site modelling by expert groups including thorough evaluation of uncertainties and the formulation of a confidence statement, the reviewing by external experts, and the pessimistic handling of many factors in the assessment all contribute to this conclusion.

Based on the sensitivity analyses of the calculated risk in SR-Site, the evaluation of candidate issues for expert elicitations is updated in Section 13.10.2, forming input to future handling of uncertain issues of importance for long-term safety.

2.9 Quality assurance

SKB applies a management system that fulfils the requirements of ISO 9001:2000 and that has been certified by DNV Certification AB, Sweden. In accordance with SKB's procedures for project management, described in the management system, quality assurance plans for the Spent fuel project (Kärnbränsleprojektet), of which SR-Site is a sub-project, and for the SR-Site project have been developed and implemented. The QA plan for the SR-Site project (SDK-003) builds on the QA-plan developed for the SR-Can project /SKB 2006a/, but it has been extended and adopted to fulfil the general requirements on a project supporting the application to build a final repository, as specified in the QA plan for the Spent Fuel project (SDK-001). Other sub-projects of the Spent Fuel project have their own specific quality assurance plans, steered by the general requirements in the QA plan for the Spent fuel project. In addition, all sub-projects that provide input to the SR-Site project are required to fulfil the requirements on quality assurance as specified in the SR-Site QA plan.

The following text is based on the introductory sections of the SR-Site QA plan.

2.9.1 General

In broad terms, a QA plan for a long-term safety assessment of a spent nuclear fuel repository aids in assuring that all relevant factors for long-term safety have been appropriately included and handled in the safety assessment. Although no QA system will rigorously prove that this is the case, a purpose-designed QA plan and QA system will assist the implementer in carrying out the safety assessment in a structured and comprehensive manner and aid a reviewer in judging the quality and comprehensiveness of the assessment. Quality assurance measures are one important element in building confidence in the fact that e.g. a model representation of a site or of a process is an adequate and sufficient representation of reality for the purposes of the safety assessment.

A principal purpose of a safety assessment of a final repository is to investigate whether the repository can be considered radiologically safe over time. In principle, this is established by comparing estimated releases and associated radiation doses and risks with regulatory criteria.

A large number of factors affecting long-term safety need to be handled in the assessment in a quality-assured manner. These factors – or features, events and processes, FEPs – are collected in a database that is also used as a QA instrument. The FEP database and underlying reports demonstrate how specific FEPs are included in the assessment or why they have been excluded.

The handling of many of the FEPs occurs in modelling of the repository evolution. This requires a scientific evaluation of the understanding of the processes involved in the modelling, the formulation of mathematical models that simulate the process or system of coupled processes based on the understanding of the phenomena, the translation of the mathematical model into a computer code, the derivation of input data and execution of the code. All these aspects need to be documented and quality assured.

In establishing the SR-Site quality assurance plan, the ISO 10005 standard "Quality management – guidelines for quality plans" has been use as an overall guide.

2.9.2 Objectives of the QA plan

The objective of the QA plan is to ensure that all relevant factors for long-term safety have been appropriately included and handled in the safety assessment SR-Site. In particular, it has been designed to aid in demonstrating:

- That all factors relevant for long-term safety occurring in earlier versions of SKB databases and in the international NEA FEP database have been considered in the assessment.
- That the exclusion of any of these factors is well motivated by an identifiable expert.
- That the handling of included factors is well justified by identifiable experts.
- How quantitative aspects of the assessment are handled by mathematical models and how the models (computer codes) and the uses of the models have been quality assured.
- How appropriate data for quantitative aspects of the assessment have been derived and used in the assessment in a quality assured and traceable manner.
- How data, models and analyses used in earlier assessments are qualified for use in SR-Site, where applicable.
- How the safety assessment reports have been properly reviewed and approved for correct and complete content.

2.9.3 SR-Site steering documents

A number of steering and QA-related documents for the SR-Site project are established in the QA plan. These documents are listed in Table 2-2 and are available in SKB's internal documentation system SKBdoc.

The project was defined by issuing, within SKB, a project decision, (item 1 in Table 2-2) describing the purpose of the project, its deliverables, its time frame, necessary prerequisites in terms of data deliveries from related projects, the actors involved in the project and their roles etc.

A project plan (item 2 in Table 2-2) was established, giving more detailed descriptions of how the purposes of the project are to be fulfilled. Associated with the project plan is a risk analysis document (item 3 in Table 2-2), identifying critical issues that could jeopardise the fulfilment of the project objectives, a time plan (item 6 in Table 2-2), covering project activities and their interdependencies, and a QA plan (item 5 in Table 2-2) that builds on the general QA plan for the Spent fuel project (item 4 in Table 2-2). The remaining documents listed in Table 2-2 are associated with the SR-Site QA plan.

Table 2-2. Steering and QA-related documents for the SR-Site project.

Item	Object	Language
1.	Project decision	Swedish
2.	Project plan	Swedish
3.	Project risk analysis	Swedish
4.	SDK-001 Quality assurance plan Spent fuel project	English
5.	SDK-003 Quality assurance plan SR-Site	English
6.	Time plan	English
7.	List of experts	English
8.	Review plan for SR-Site reports	English
9.	Template for review comments	English
10.	Instructions for development and handling of the SKB FEP database – version SR-Site	English
11.	Instructions for developing process descriptions in SR-Site and SR-Can	English
12.	Instruction for qualification of "old" references	English
13.	Plan for model and data quality assurance for the safety assessment SR-Site	English
14.	SR-Site Model summary report instruction	English
15.	Instruction for supplying data for the SR-Site data report	English
16.	Instruction for task descriptions for the safety assessment SR-Site	English
17.	Instruction for final control of data used in SR-Site calculations/modelling	English

2.9.4 Expert judgements

Information based on expert judgements of various kinds permeates safety assessments. Expert judgements can include anything from a scientist's interpretation of a result of a straightforward experiment to an expert's judgement on the impact of human-induced greenhouse effects on the future climatic evolution or to an assessment of the likelihood of the occurrence of a particular process/phenomenon that could have an impact on repository evolution. A judgement could consist of anything from a well-justified quantitative or qualitative statement in a report, to an exhaustive and formal questioning of a carefully selected panel of experts using an approved elicitation protocol.

Furthermore, there are issues on which different experts have differing views. In cases where a consensus view or statement cannot be achieved, it is necessary to take the differing views into account in the assessment. This can be done e.g. through the formulation of several calculation cases or by choosing the most pessimistic approach, if such an approach can be shown to exist.

Most of the information based on expert judgements in SR-Site is provided either in the form of reports written by one or several experts or as decisions made by generalists in e.g. the screening of FEPs, the selection of scenarios or the formulation of calculation cases. Formal questioning of a panel of experts has not been employed in SR-Site, see further Section 2.8.5.

Documentation of expert judgements

For the traceability of the assessment, it is important to clearly state where expert judgements are made and by whom. For this purpose, all important experts providing, in one way or another, expert judgements for SR-Site are listed in a separate document (item 7 in Table 2-2). This list contains information on the role of the experts in SR-Site, a motivation for their selection as experts and references to where at SKB the credentials of the experts are filed. The list of experts will not be published, but will be made available to reviewing authorities on request.

References to the appropriate experts are then made where the judgements are used in the assessment. In order not to burden the text with numerous references of this nature, these are, for this main report, provided in the preface. In several of the other central reports, e.g. the **Process reports** and the **Data report**, relevant references are made to these experts. For the underlying reports, the experts are readily identified as the author(s) of the report in question.

Also, the present and other reports have been subject to scrutiny within SKB and by external reviewers, as documented in the review protocols (see Section 2.9.5). These reviewers are also included in the list of experts.

Selecting experts

In general no formal rules for the selection of experts have been applied. The generalists in the core team of the assessment provide a large fraction of the expert judgements and these individuals have been working with the safety of the KBS-3 system for a number of years and are, therefore, among the most experienced individuals available on the various aspects of the analysis of the system. This does, however, also imply a risk of bias, stressing the importance of external reviews of the material developed within the project, see further Section 2.9.5.

Regarding experts for the documentation of process understanding, for the selection of models or of input data for the quantitative aspects of the assessment, the ambition has been to contract leading experts in the field. These reviewers should be recognised experts (i.e. qualified and experienced, as reflected by their research publication record) in the relevant scientific field. The merits of these experts are documented, providing a sufficient justification for their involvement.

2.9.5 Peer review

Peer reviewing with subsequent handling of review comments is an important method for broadening the basis on which expert opinions/judgements are formed in a safety assessment. All reports produced in the SR-Site project have therefore been subject to peer review prior to being finalised, according to a review plan established for the project (item 8 in Table 2-2). This review plan defines the document that should be provided to the reviewers, general criteria for acceptance of a report, requirements on reviewers' competence and how the review documents shall be handled. This review plan further prescribes that a review instruction is produced for each report subject to review in which the general as well as report-specific acceptance criteria are specified together with the selected reviewers and their competence.

A template for review comments has been used (item 9 in Table 2-2) and this template also requires the author of the report to document how each comment is handled when the report is finalised. These review statements are filed in SKB's internal documentation system SKBdoc to ensure traceability of the review process.

A requirement on all documents that are part of the license application is that they have been subjected to a factual and quality review. However, many of these new documents need to refer to older SKB, or SKB external, documents that lack a documented factual and quality review. For example, process descriptions in the SR-Site **Process reports** contain many references to old or external documents of this type. A procedure for qualification of documents that are used as references in SR-Site reports has, therefore, been established (item 12 Table 2-2). This procedure is restricted to documents regarded as supporting references, i.e. documents that are used to support or justify a decision, selection or treatment of an issue in SR-Site. The procedure implies that any results or arguments adopted from supporting references, at least from those lacking a documented factual and quality review, should be given in such detail that it is possible to assess the reliability in the argumentation without having to consult the supporting references. Furthermore, the reasons why a supporting reference is judged as adequate from a quality assurance perspective are documented in the process descriptions in the **Process reports** and the **Climate report** (see Section 7.3). The argumentation from and qualification of the supporting references has then been reviewed by the experts selected for factual review of the reports in question, e.g. the **Process reports**.

3 FEP processing

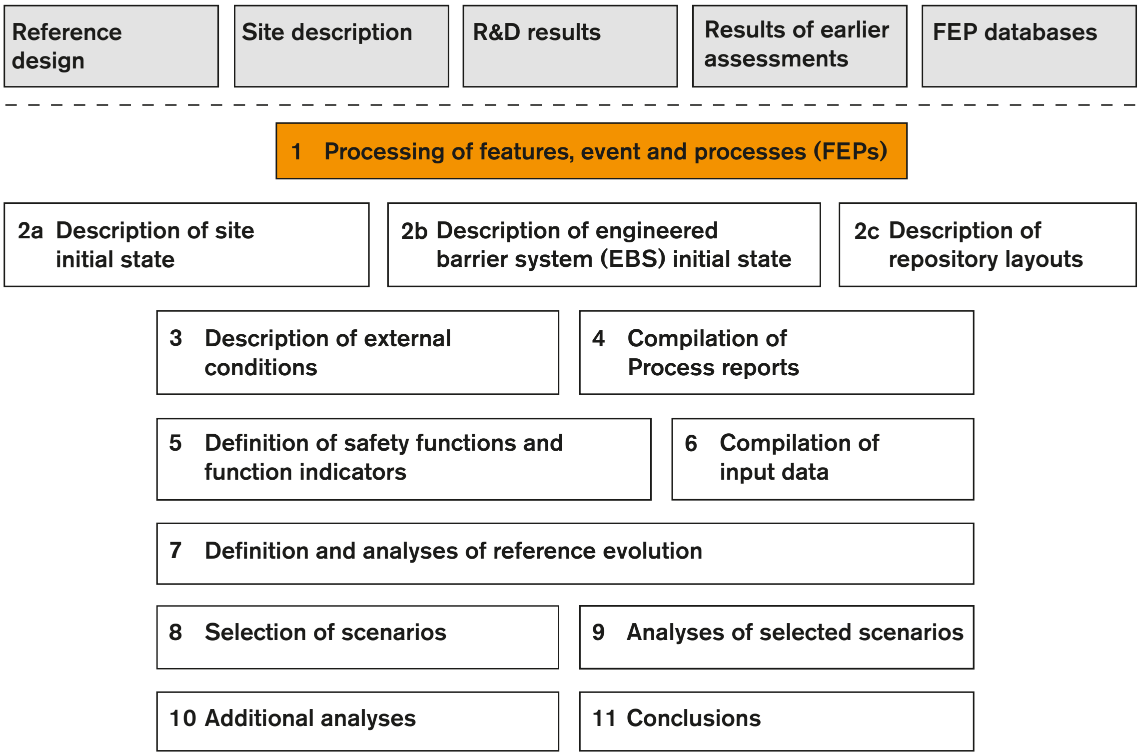


Figure 3-1. *The SR-Site methodology in eleven steps (Section 2.5), with the present step highlighted.*

3.1 Introduction

Much of the methodology described in the previous section is related to the handling of FEPs (Features, Events, Processes) through the different steps of the safety assessment. This section describes in more detail how FEPs are handled throughout the assessment and the various tools used to ensure systematic and comprehensive handling.

A main aim in the FEP handling was to establish a catalogue of FEPs that needed to be addressed in the SR-Site assessment.

3.2 SKB FEP database

An important and formal tool for ensuring that all relevant factors have been considered in the safety assessment is provided by available databases of features, events and processes (FEPs) relevant to long-term safety of nuclear waste repositories. An SKB FEP database has been developed for SR-Site, as described in the **FEP report**. This FEP database builds on the outcome of the FEP work conducted in SKB's two most recent major safety assessments, the SR-Can and SR 97 assessments, as reported in the SR-Can FEP report /SKB 2006b/, the SR 97 Process report /SKB 1999b/ and the supporting documentation on the Interaction matrices developed for a deep repository of the KBS-3 type /Pers et al. 1999/.

In the SR 97 Process report /SKB 1999b/, comprehensive sets of long-term processes relevant to repository safety for each of the system components, i.e. fuel, canister, buffer/backfill and geosphere, were identified. For each component, a set of variables needed to describe the evolution of the state of the component over time was also established. As a first step in the development of the SKB FEP database, these identified processes and variables were collected in an SR 97 FEP database, forming an important starting point for the SR-Can FEP handling.

The SR 97 database was then systematically compared with other national databases included in version 1.2 of the NEA international FEP database /NEA 1999b/, to ensure that all relevant factors were taken into account. In version SR-Can of the SKB FEP database, all items were classified as one of the following:

- Processes within the system boundaries relevant to long-term safety and the system component specific variables required to describe the state of the component at a specified point in time.
- Factors affecting the initial state of the repository, either directly related to a specific aspect or to the initial state in general.
- External factors relevant to long-term safety, e.g. evolution of climate and climate related issues, and human intrusion.

Most FEPs in version 1.2 of the NEA database could be mapped to one of these categories. All other FEPs were characterised as general methodology issues or determined to be irrelevant for the KBS-3 system.

Since the completion of the FEP work within SR-Can, an updated electronic version, version 2.1, of the NEA FEP database has become available /NEA 2006/. Compared to version 1.2 of the NEA FEP database, version 2.1 contains FEPs from two more projects. As part of SR-Site, all new project FEPs in version 2.1 of the NEA FEP database have been mapped according to the methodology adopted in SR-Can resulting in an SR-Site version of the SKB FEP database. The SKB FEP database thus encompasses the SR 97 version, the SR-Can version and the SR-Site version of the FEP database. The SR 97 version contains the SR 97 processes and variables. The SR-Can and SR-Site versions contain all FEPs in the NEA database and in the national databases linked to the NEA database, versions 1.2 and 2.1, respectively, including the classification and characteristics of these FEPs. A more detailed description of the SKB FEP database, in particular the SR-Site version, is provided in the **FEP report**.

3.3 SR-Site FEP catalogue

Based on the FEP processing conducted for SR-Can and briefly described above, an SR-Can FEP catalogue was established. This FEP catalogue contains all FEPs that needed to be handled in SR-Can and is thus fundamentally a subset of FEPs in the SKB FEP database. However, some of the system components were not treated in detail in SR-Can, and for these system components preliminary FEPs were included in the SR-Can FEP catalogue. These components were tunnel plugs, backfill materials for cavities other than the deposition tunnels, the bottom plates in the deposition holes and borehole seals. Furthermore, biosphere processes were not included in the SR 97 Process report and there was, therefore, not the same basis for updating these descriptions in SR-Can as for the engineered barriers and the geosphere. Therefore, the SR-Can FEP catalogue contains provisional biosphere FEPs corresponding to the sub-components of the biosphere system, to which biosphere FEPs in the NEA FEP database are mapped.

The SR-Site FEP catalogue is a developed version of the SR-Can FEP catalogue. For the system components not treated in detail in SR-Can, as well as for the biosphere, SR-Site FEPs have been defined and included in the FEP catalogue. The mapping of NEA Project FEPs made to the preliminary and provisional FEPs for these system components in SR-Can has been revisited and a new mapping has been made to the FEPs now included in the SR-Site FEP catalogue. The categories of FEPs in the SR-Site catalogue are listed below.

- Initial state FEPs.
- Processes in fuel, canister, buffer, backfill, tunnel plug, central area, top seal, bottom plate in deposition holes, borehole seals and geosphere.
- Variables in fuel, canister, buffer, backfill, tunnel plug, central area, top seal, bottom plate in deposition holes, borehole seals and geosphere.
- Biosphere FEPs.
- External FEPs.
- Methodology issues.

In addition, there is a possibility to enter in the FEP catalogue any issue that is, for whatever reason, identified as relevant for the safety assessment. For SR-Can, some site-specific issues identified in the preliminary safety evaluation of the sites were included. For Forsmark these issues concerned the potential impact of nearby nuclear power plants and the power cable to Finland and the effect of a deep mine excavation near, but outside, the tectonic lens at Forsmark. For SR-Site, no additional issues that are not covered by FEPs already included in the SR-Site FEP catalogue have been identified.

In the following, each category is briefly described.

Initial state FEPs

This category describes deviations from the intended initial state as a consequence of undetected mishaps, sabotage, repository left open, etc. These are propagated to the selection of scenarios described in Chapter 11. The initial state FEPs in the SR-Site FEP catalogue are in essence the same as those in the SR-Can FEP catalogue. The only exception is that two initial state FEPs defined for backfill of other repository parts in the SR-Can catalogue, in the SR-Site catalogue are replaced by two initial state FEPs defined for the central area and two for the top seal.

It should be noted that the intended initial state with tolerances, the reference initial state, is one of the bases for the main scenario. The reference initial state for the different system components is described in the **Spent fuel report**, the **Canister-**, **Buffer-**, **Backfill-** and **Closure production reports** and in the **Underground openings construction report**, and summarised in Chapter 5. In the FEP catalogue, each variable record, see below, contains also a reference to the description of the reference initial state for that variable.

Processes

These FEPs are long-term processes relevant to repository safety for each of the system components fuel, canister, buffer, backfill, tunnel plug, central area, top seal, bottom plate in deposition holes, borehole seals and geosphere. All internal processes are comprehensively documented in a number of **Process reports**, see further Chapter 7. The handling of all processes in the fuel, canister, buffer, backfill and geosphere is summarised in process tables, given in Chapter 7. There are typically around 20 processes for each system component.

A few modifications in the list of internal processes for the system components fuel, canister, buffer, backfill and geosphere have been made compared to the list of processes included in SR-Can. These modifications were not initiated by the complementary mapping of new project FEPs in version 2.1 of the NEA FEP database, but have been made to improve the structure and logic of the descriptions. For example, to improve the handling of uncertainties in the geochemical evolution of the buffer, some mechanisms included in integrated descriptions in SR-Can are in SR-Site included as separate processes, e.g. iron-bentonite interactions and cementation. Another example concerns a modification in the list of geosphere processes. In SR-Can, surface erosion and weathering was described in the Geosphere process report /SKB 2006d/, but in SR-Site the description of these mechanisms is included in the **Climate report** and also addressed and considered in the biosphere analyses and reporting. For the system components not treated in detail in SR-Can, i.e. tunnel plugs, central area, top seal, bottom plate in deposition holes and borehole seals, process FEPs have been established largely based on the list of processes defined for the system components buffer and backfill. The outcome of the mapping in SR-Site of FEPs in the NEA FEP database was used to check that no relevant processes are missing in the set of processes for these system components.

Variables

These FEPs are the variables needed to describe the evolution of the state of the fuel, canister, buffer, backfill, tunnel plugs, central area, top seal, bottom plate in deposition holes, borehole seals and geosphere over time. They are thus essentially tables with definitions. The identification of variables has been done by the experts responsible for the documentation of the processes relevant for long-term safety. The sets of variables were established in conjunction with the documentation of the processes, since it had to be ensured that the variable sets were suited to describe all conceivable alterations of the barrier properties as a result of the long-term processes. There are typically around 10 variables for each system component. In the FEP catalogue, each variable record contains a reference to the description of the reference initial state for that variable.

The handling of influences between processes and variables is described in Section 3.4.

Biosphere FEPs

Biosphere processes were not included in the SR 97 Process report /SKB 1999b/ and there was not the same basis for updating these descriptions in SR-Can as for the engineered barriers and the geosphere. In SR-Can, provisional biosphere FEPs were defined and included in the SR-Can FEP catalogue. For SR-Site, a biosphere process report has been developed. That report contains general descriptions of the processes considered to be of importance for the safety assessment, whereas the site-specific aspects of the processes and how these are handled in the safety assessment are provided in the various ecosystem reports developed for SR-Site /Andersson 2010, Aquilonius 2010, Löfgren 2010/, see Section 7.1.2. In the SR-Site FEP catalogue, a FEP record is included for each biosphere process defined in the biosphere process report. These FEP records contain references to the corresponding process description in the biosphere process report as well as to the report where the handling of the process in SR-Site is documented.

External FEPs

FEPs in the NEA database defined as external FEPs in SR-Can were subdivided into the categories listed below. The complementary mapping of new project FEPs in the NEA FEP database version 2.1 carried out for SR-Site has not pointed out any need to modify the categorisation of external FEPs. Consequently, the categories of external FEPs in SR-Can and in SR-Site are:

- climate related issues,
- large-scale geological processes and effects,
- future human actions,
- other.

Climate issues and their handling are described in the **Climate report**. In SR-Can, ten climate FEPs were defined to represent the climate issues described in the SR-Can Climate report /SKB 2006c/. The NEA Project FEPs associated with these SR-Can climate FEPs and their handling in SR-Can were documented in the SR-Can FEP catalogue. The SR-Site FEP catalogue contains essentially the same climate FEPs as the SR-Can FEP catalogue, although somewhat restructured, but also one additional FEP, “Denudation”. This additional FEP corresponds to the SR-Can geosphere process “Surface weathering and erosion”, which in SR-Site has been categorised as a climate related and biosphere issue rather than a geosphere process. In SR-Site, the handling of each NEA Project FEP has been revisited and updated as appropriate, including new NEA Project FEPs in version 2.1 of the NEA FEP database that are mapped to these climate FEPs.

Large-scale geological processes and effects were covered in SR-Can by two FEPs in the SR-Can FEP catalogue and described in the SR-Can Geosphere process report /SKB 2006d/. As with the climate-related issues, it was checked that relevant aspects of the NEA Project FEPs mapped to these large-scale geological process FEPs were covered in the descriptions of these processes in the Geosphere process report. The outcome of this check was documented in the SR-Can FEP catalogue. In SR-Site, the documentation in the SR-Can FEP catalogue has been revisited and updated as appropriate, considering also new NEA Project FEPs in version 2.1 of the NEA FEP database that are mapped to these large-scale geological process FEPs.

Future human actions and how these are handled in the safety assessment are described in the **FHA report**. In SR-Can, seven FHA FEPs were defined to represent future human actions described in the SR-Can FHA report /SKB 2006e/. Each NEA Project FEP mapped to these SR-Can FHA FEPs and their handling in SR-Can was documented in the SR-Can FEP catalogue. The same seven FHA FEPs are included in the SR-Site FEP catalogue. However, the documentation of the handling of each NEA Project FEP mapped to these SR-Site FEPs has been revisited and updated as appropriate, considering also new NEA Project FEPs in version 2.1 of the NEA FEP database that are mapped to these SR-Site FHA FEPs.

In the category “other”, only meteorite impact was identified in SR-Can, but excluded from further analysis, see Section 6.1. However, meteorite impact was still defined as a FEP in the SR-Can FEP catalogue for documentation purposes and the justification for excluding this FEP from further analysis was documented in the FEP record in the SR-Can FEP catalogue. The audit of the new NEA Project FEPs in version 2.1 of the NEA FEP database has not indicated any need for modifications. Therefore, for documentation purposes, the FEP is maintained in the SR-Site FEP catalogue.

The handling of external FEPs is further addressed in Chapter 6.

Methodology issues

A number of relevant issues relating to the factual basis for the assessment and to the methodology of the assessment were identified in the NEA FEP database. Most of these are of a very general nature, but were for the sake of comprehensiveness also included in the SR-Can FEP catalogue. The audit of the new NEA Project FEPs in version 2.1 of the NEA FEP database has not resulted in any need for modifications. Therefore, the SR-Site FEP catalogue contains the same methodology FEPs as the SR-Can FEP catalogue.

3.4 Couplings

FEPs are coupled in several ways and on several levels. Couplings between processes and variables occur within a system component and the system components influence each other in several ways. The following is a description of different types of couplings and tools used to document and visualise them.

Influence tables

Within a system component, each process is influenced by one or several of the variables describing the state of the component and the process, in turn, influences one or several of the variables. These couplings within a system component are described by influence tables, one for each process in the **Process report**. A distinction is made between influences that exist in principle but are sufficiently insignificant to be neglected in the safety assessment and those that require a detailed treatment. The handling of the latter category is explicitly mentioned as the handling of the process in question is established in the **Process reports**. See Section 7.3 for an example of an influence table. The influence tables are fed back to the FEP catalogue from the **Process reports**. Couplings across system components are described in the process descriptions in the **Process reports** under the heading “Boundary conditions”, see further Section 7.3.

Process diagrams

Based on the sets of variables and processes and the influences between them, a process diagram can be constructed for each system component. This is done in the FEP database. The diagram essentially takes the form of a table with the processes as lines and the variables as columns. The table matrix consists of arrows visualising the presence of influences between processes and variables and a colour coding of the arrows displaying if the influence is handled or not in the assessment.

AMF, Model summary report

When evaluating repository evolution, a number of coupled or interacting models are utilised. This set of models and the dependencies/interactions between them are described by two assessment model flow charts, AMFs. One concerns the excavation/operation and initial temperate period and one represents permafrost and glacial conditions. These are further described in Section 7.5. A **Model summary report** describes the models represented in the AMFs. The AMFs are also included in the FEP database as well as the process tables and the tables that links the AMF and process tables, see Sections 7.4 and 7.5, respectively.

FEP chart

On a higher level, it is desirable to have an instrument providing an overview of how critical initial state factors, variables, processes and external factors influence the safety functions of the repository. To serve this purpose, a FEP chart has been developed. This is further described in Section 8.5, following a discussion on repository safety and the definition of a number of safety function indicators. The FEP chart is also included in the FEP database with links to the process table that also is included in the FEP database.

Handling of couplings

The overwhelming majority of the large number of couplings that are in principle present in the system are not directly included in any of the models used to quantify the development of the system. Justification for the neglect of couplings is provided e.g. in the **Process reports** where i) some processes, and thereby also their couplings, are argued to be negligible, and ii) where the inclusion or neglect of internal couplings is justified in conjunction with the influence tables, see Section 7.3.

Furthermore, in many cases, a modelling effort results in a bounding value on a certain property of the system, and this result is then used to put bounds on related phenomena of relevance for safety, without a detailed coupled model. The near-field temperature is an example: By modelling, it is demonstrated that the buffer temperature never exceeds a certain limit and this result is used to argue that long-term mineral alterations in the buffer can be neglected, i.e. the coupling between temperature and chemical reactions is addressed without a fully coupled model. In this example, a further detail concerns the handling of the thermal properties of the buffer for which it is assumed that the buffer is unsaturated, which maximises the calculated peak temperatures, i.e. the coupling between buffer saturation and temperature is taken into account from the point-of-view of safety without a coupled model. Many of these relationships are demonstrated in the process tables, Table 7-2 to Table 7-6 in Chapter 7.

The modelling and couplings are thus not in all respects aiming at a realistic representation of the system, but at one that is adequate for the argumentation in the safety assessment, an approach that in many cases allows considerable simplifications to be made.

The detailed justification of the approach is provided through the collected information behind modelling approaches and assumptions regarding all the particular issues, informed by the understanding of the safety relevant aspects of the system.

The handling of FEPs in SR-Site is summarised in Figure 3-2.

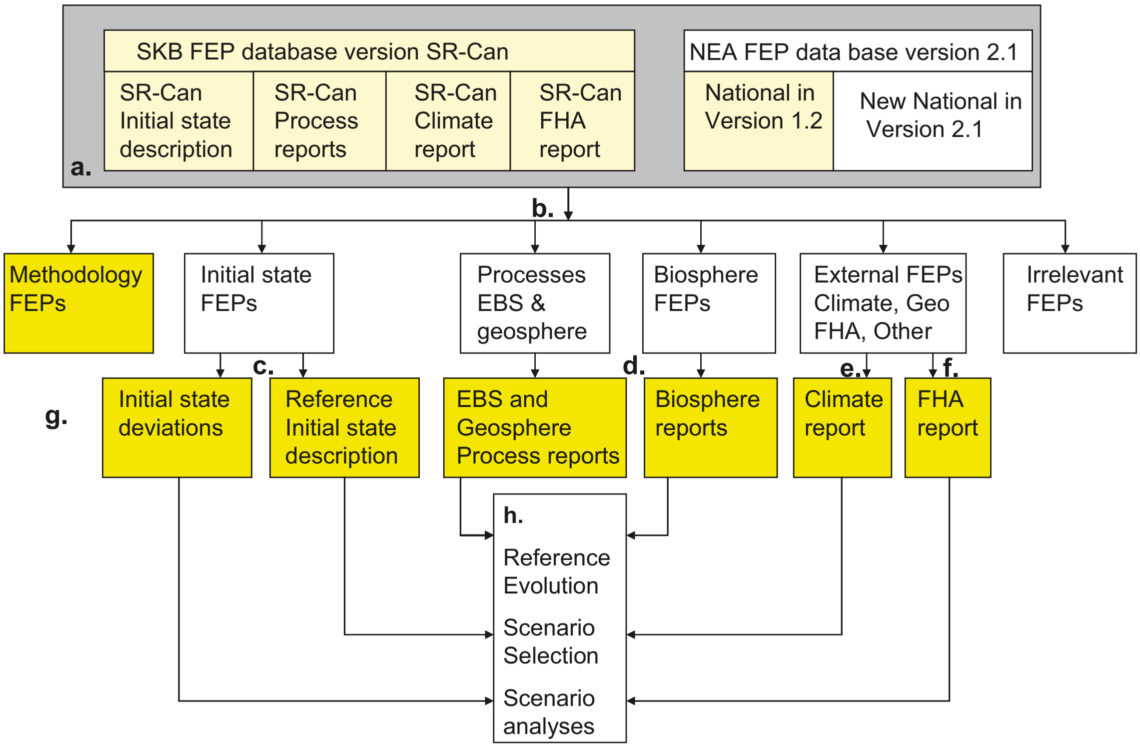


Figure 3-2. *The handling of FEPs in SR-Site.*

a. The starting points for the SR-Site FEP handling are FEPs in the SR-Can version of the SKB FEP database including the SR-Can FEP catalogue and associated SR-Can reports, and the two national data bases that are new in the NEA international FEP database version 2.1 as compared with version 1.2, which was the starting point for the SR-Can version of the SKB FEP database.

b. FEPs are sorted into three main categories: i) initial state, ii) internal process and iii) external FEPs. FEPs are also categorised as irrelevant or as being related to methodology at a general level.

c. Initial state FEPs are either i) included in the initial state description in SR-Site, i.e. the reference description of the KBS-3 repository, the site description or the site-specific layout of the repository or ii) categorised as initial state deviations to be further handled in scenario selection.

d. Process FEPs are used to update the SR-Can set of internal processes for the EBS (Engineered Barrier System) and the geosphere. The resulting SR-Site set of processes are documented in the SR-Site **Process reports**. Biosphere FEPs are defined in the **Biosphere process report** and the handling of each process is described in various biosphere reports.

e. The handling of external FEPs related to long-term climate changes is documented in the SR-Site **Climate report**. The few external, large-scale geosphere FEPs are addressed in the **Geosphere process report**.

f. The handling of external FEPs related to future human actions (FHA) is developed in the SR-Site **FHA report**. The only "other" external FEP, meteorite impact, was dismissed in SR-Can as being extremely unlikely and this applies also for SR-Site. No new "other" external FEPs have been identified for SR-Site.

g. The FEPs handled in the yellow boxes constitute the SR-Site FEP catalogue.

h. The reference initial state, all long-term processes and a reference external evolution are used to define a reference evolution for the repository system. This evolution is an important basis for specifying a comprehensive main scenario. A set of additional scenarios address e.g. deviations from the reference initial state and from the reference external evolution as well as situations related to FHA.

4 The Forsmark site

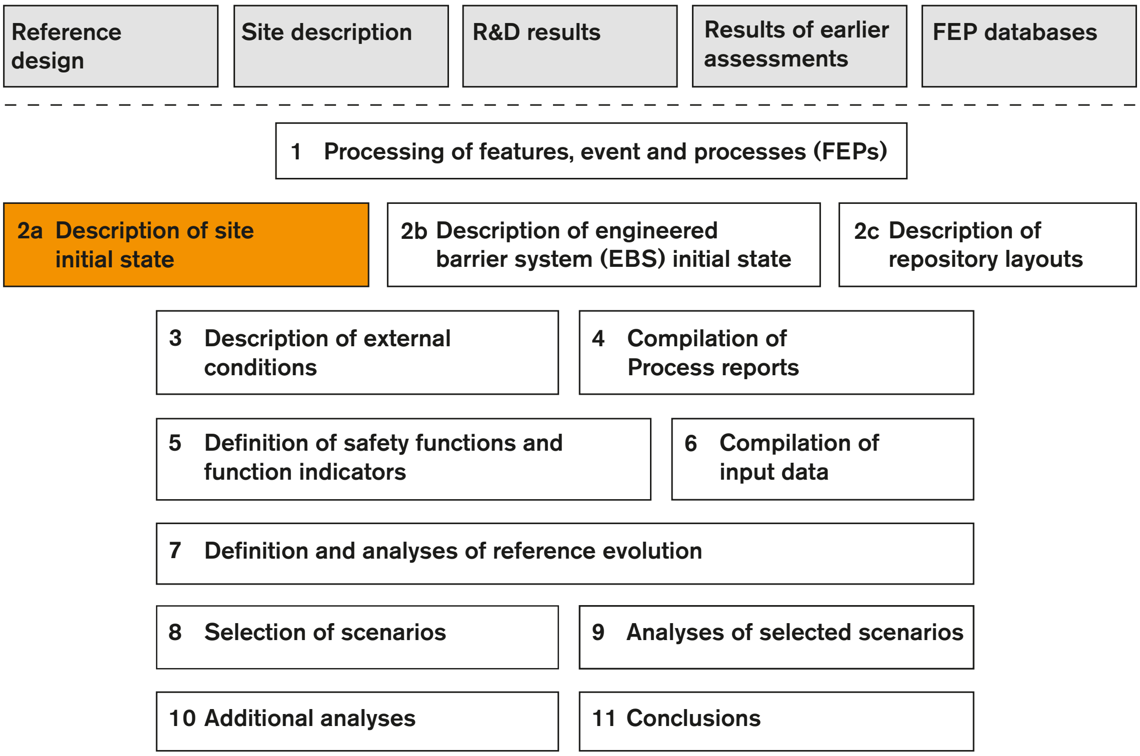


Figure 4-1. *The SR-Site methodology in eleven steps (Section 2.5), with the present step highlighted.*

4.1 Introduction

A comprehensive description of the initial state of the repository system is one of the main bases for the safety assessment (Section 2.5.2), and is also the basis for development of repository design. For the geosphere and the biosphere, the state at the time of beginning of excavation of the repository is a natural starting point, since the knowledge of this relatively undisturbed state is available through the site-descriptive models that are derived from site investigation data. Furthermore, the short-term evolution of the host rock from the undisturbed state to that after excavation has to be considered in a safety assessment that is based on observations made prior to excavation.

Based on these considerations, the initial state of the geosphere and biosphere in SR-Site is defined as the natural, undisturbed state at the time of beginning of excavation of the repository, see further Section 5.1. A repository design, including a site-specific layout, is also developed based on the undisturbed state documented in the site description, see Section 5.2. Short-term geosphere processes or alterations due to repository excavation are documented in the **Geosphere process report**, see Chapter 7, and in the **Underground openings construction report**, Section 5.2. This means that the evolution of the natural system, including the potential evolution of an excavation damaged zone, EDZ, will be followed from the time of beginning of excavation of the repository, see Section 10.2.

The information transfer from the site investigations at Forsmark to the safety assessment application has involved several steps.

- Field data have been obtained from various investigation activities like airborne and ground geophysics, borehole drilling and borehole testing. After quality control, the data have been entered into the SKB data bases Sicada and GIS.
- The field data have been interpreted and evaluated into a cross-disciplinary site descriptive model (SDM), being a synthesis of geology, rock mechanics, thermal properties, hydrogeology, hydrogeochemistry, bedrock transport properties and surface system properties, see Figure 4-2. The SDM provides a description of the understanding of the site properties within the different

disciplines and it also provides an assessment of the uncertainty in these descriptions. The SDM for the Forsmark site at the completion of the surface-based investigations is reported in a main site-description report, **Site description Forsmark**, and a number of supporting reports.

- **Site description Forsmark** has been used to develop a site-specific design of the repository /SKB 2009b/, in accordance with stated design premises. This is described in the **Underground openings construction report**, and discussed in Section 5.2 of this report.
- The site description and references therein cannot always be used directly in the safety assessment. There is a need to also consider non-site specific information, to add judgements on how to handle the uncertainties identified in the site description and to make final selections of model input data. For this reason, all site data used in SR-Site are assessed in the **Data report**, using the SDM as input. The role of the **Data report** is explained in Section 2.5.6 and the format of the **Data report** is further discussed in Chapter 9.

In short, the quantitative site-specific input to SR-Site is handled in the **Data report**, and is not repeated here. There is, however, a need to outline the main characteristics and current understanding of the site. The remainder of this chapter provides an overview with this in mind, largely extracted from the site synthesis provided in Chapter 11 of the **Site description Forsmark**. The overview of the surface system provided in Section 4.10 is, in addition to the **Site description Forsmark** report, based on the surface system description report /Lindborg 2008/, one of the main background reports to the **Site description Forsmark**.

As part of the site-descriptive modelling, the uncertainty and confidence in the Forsmark site description were assessed. This assessment comprised exploring confidence in the site characterisation data, key remaining uncertainties in the site description, alternative models and their handling, consistency between disciplines and the main reasons for confidence or lack of confidence in the site descriptive model. The overall outcome of this assessment was that it was found that the site properties of importance for both repository constructability and long-term safety are sufficiently bounded by quantitative uncertainty estimates or alternative models. A summary of the assessment is provided in Chapter 11 of the **Site description Forsmark** and more extensively reported in /SKB 2008b/. In the remainder of this chapter, the confidence and remaining uncertainties in the site characteristics are qualitatively addressed as part of demonstrating the current understanding of the site.

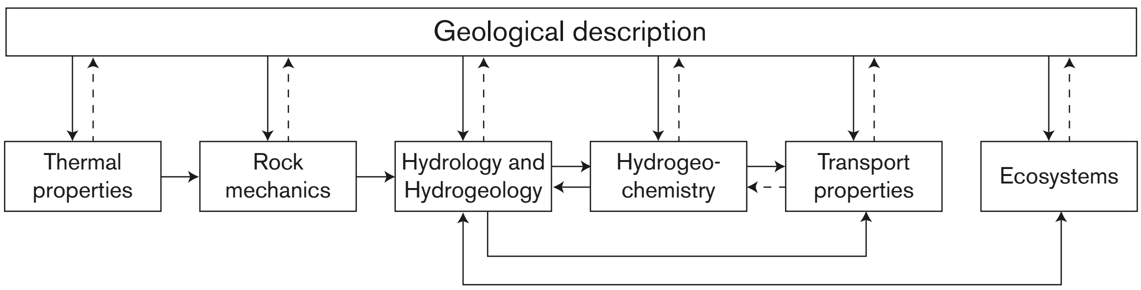


Figure 4-2. *The different discipline descriptions in the SDM are interrelated with several feedback loops and with geology providing the essential geometrical framework.*

4.2 The Forsmark area

4.2.1 Setting

The Forsmark area is located in northern Uppland within the municipality of Östhammar, about 120 km north of Stockholm (Figure 4-3). The candidate area for site investigation, approximately 6 km long and 2 km wide, is located along the shoreline of Öregrundsgrepen, a funnel-shaped bay of the Baltic Sea. The candidate area extends from the Forsmark nuclear power plant and the access road to the SFR-facility, a repository for low- and intermediate level radioactive waste, in the north-west to Kallrigafjärden in the south-east (Figure 4-3 and map in Appendix C).

The current ground surface in the Forsmark region forms a part of the sub-Cambrian peneplain in south-eastern Sweden. This peneplain represents a relatively flat topographic surface with a gentle dip towards the east that formed more than 540 million years ago. The candidate area at Forsmark is characterised by a small-scale topography at low altitude (Figure 4-4). The most elevated areas to the south-west of the candidate area are located at c. 25 m above current sea level. The whole area is located below the highest coastline associated with the last glaciation, and large parts of the candidate area emerged from the Baltic Sea only during the last 2,000 years. Both the flat topography and the still ongoing shore-level regression of c. 6 mm per year strongly influence the current landscape (Figure 4-4). Sea bottoms are continuously transformed into new terrestrial areas or freshwater lakes, and lakes and wetlands are successively covered by peat.

4.2.2 Target area for the repository

The north-western part of the candidate area has been selected as the target area for the repository (Figure 4-5 and map in Appendix C). Characterisation of the bedrock in the target area and the surroundings has been undertaken by both surface investigations and by investigations in boreholes, see Figure 4-5. Surface investigations have included geological mapping, different ground and airborne geophysical investigations, surface ecological investigations and monitoring of, for example, meteorological parameters and water levels in lakes and in the Baltic Sea. Borehole data in support of the Forsmark site description come from 25 core-drilled boreholes at 12 drill sites (Figure 4-5). These boreholes range in depth down to c. 1,000 m and have a total borehole length of c. 17,800 m. The database also contains results from investigations in 38 percussion-drilled boreholes, with a total borehole length of c. 6,500 m, and more than 100 monitoring wells in the Quaternary cover, so-called soil wells. The site information available for establishing the site description for Forsmark is fully set out in the **Site description Forsmark**.

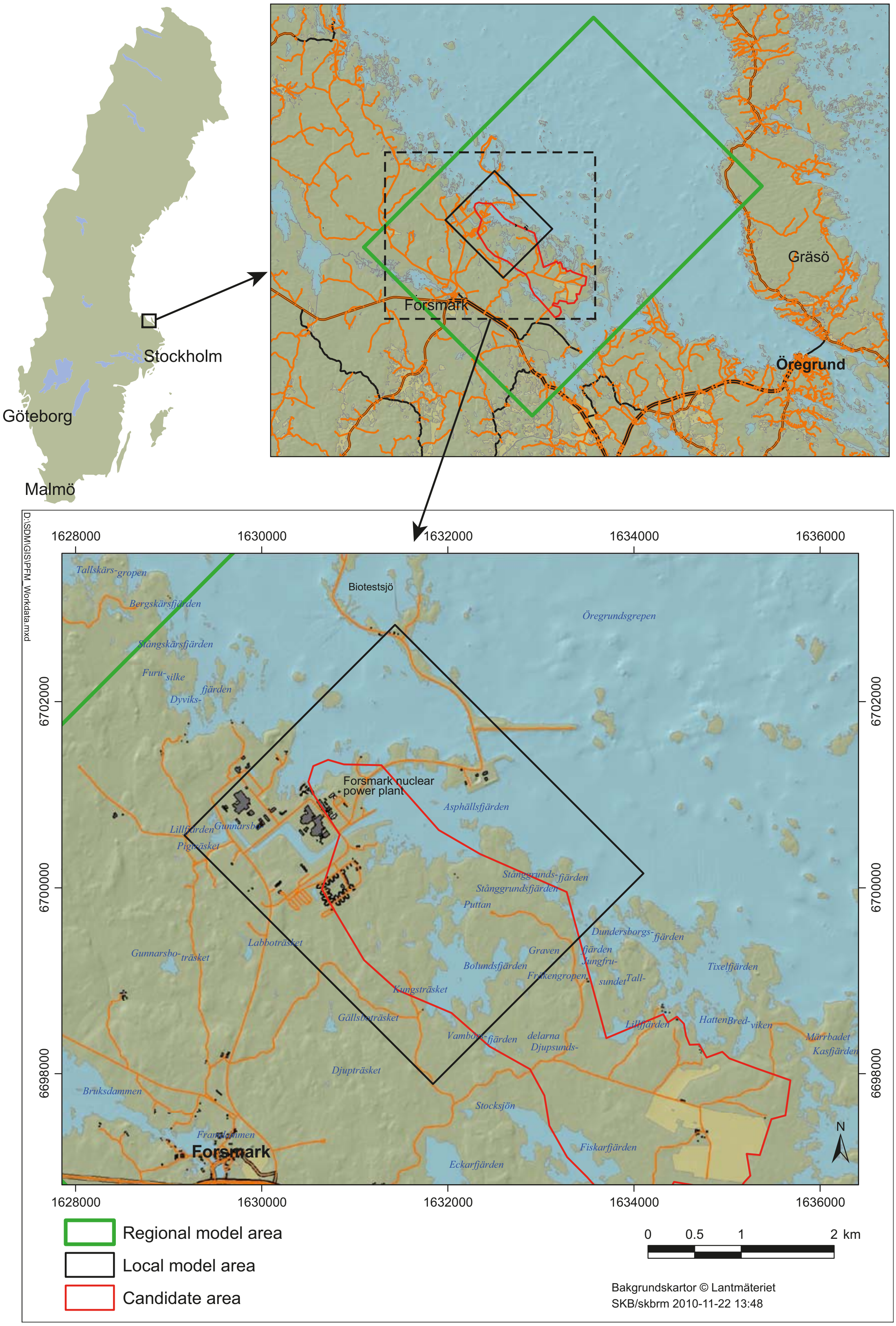


***Figure 4-3.** Location of the Forsmark candidate area (red) for site investigation.*

*Figure 4-4. Photographs from Forsmark showing the flat topography and the low-gradient shoreline with recently isolated bays due to land uplift (Figure 1-6 in the **Site description Forsmark**).*

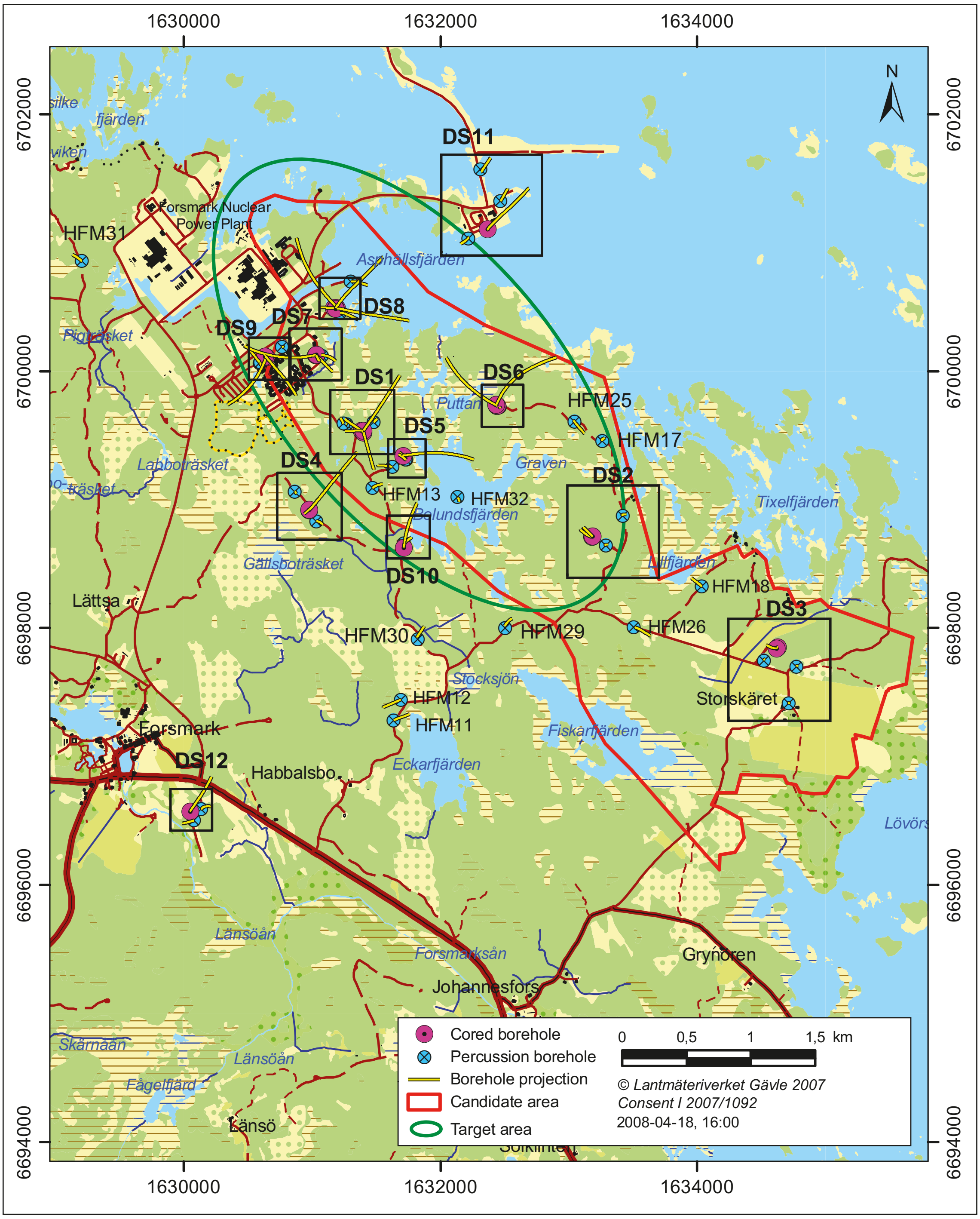


Figure 4-5. *The Forsmark candidate area with the target area in the north-western part (ringed in green) and the location of the drill sites (Figure 11-1 in the* ***Site description Forsmark****).*

4.3 Rock domains and their associated thermal and rock mechanics properties

The site lithology, i.e. the occurrence and distribution of rock types, reveals important aspects of the homogeneity of the site. Furthermore, it is directly related to the potential for mineral resources as well as thermal and rock mechanics properties of the intact rock. In the site descriptive model, the lithology is described in terms of rock domains, defined on the basis of composition, grain size, homogeneity, and style and degree of ductile deformation.

4.3.1 Rock composition and division into rock domains

The Forsmark area consists of crystalline bedrock that belongs to the Fennoscandian Shield and formed between 1.89 and 1.85 billion years ago during the Svecokarelian orogeny. The bedrock has been affected by both ductile and brittle deformation. The ductile deformation has resulted in large-scale, ductile high-strain belts and more discrete high-strain zones. Tectonic lenses, in which the bedrock is less affected by ductile deformation, are enclosed in between the ductile high-strain belts. The candidate area is located in the north-westernmost part of one of these tectonic lenses. This lens extends from north-west of the nuclear power plant south-eastwards to the area around Öregrund (Figure 4-6).

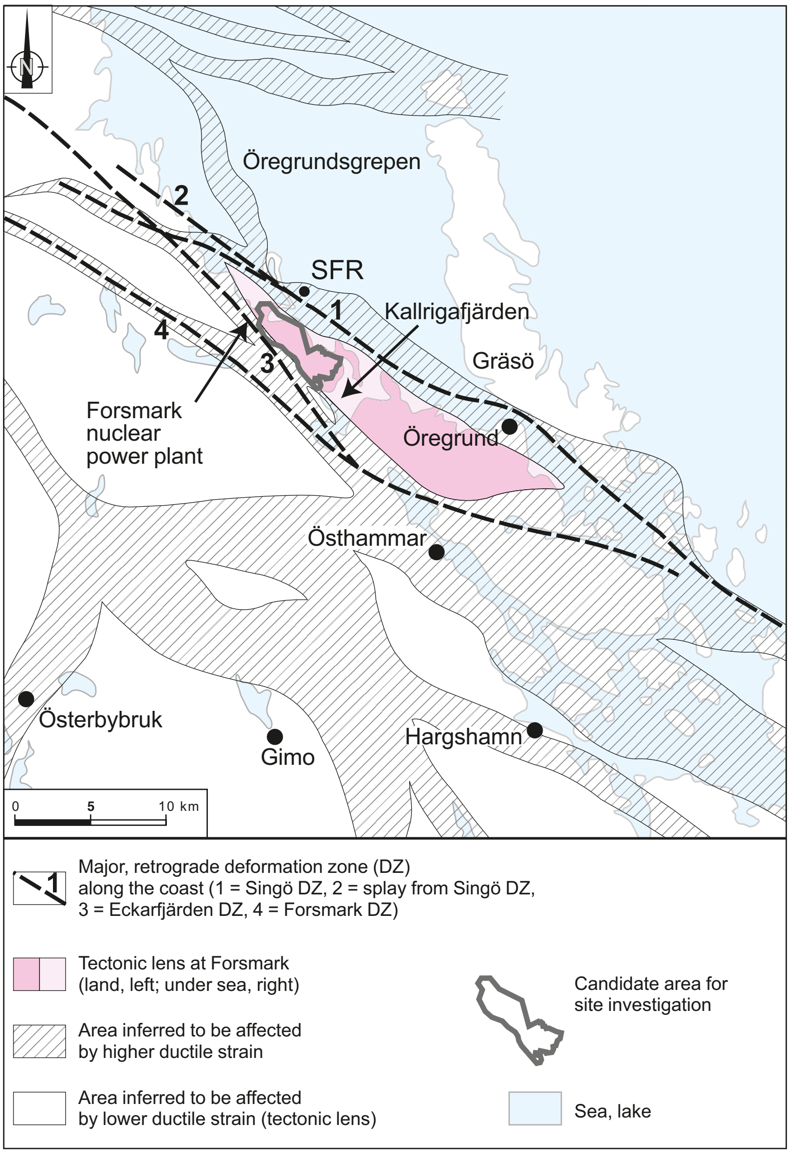


Figure 4-6. *Tectonic lens at Forsmark and areas affected by strong ductile deformation in the area close to Forsmark (Figure 1-5 in the* ***Site description Forsmark****).*

Rock domains in the target volume

Due to its internal homogeneity, most of the lens in the candidate area can be described in terms of two rock domains referred to as RFM029 and RFM045 (Figure 4-7). These are also the two rock domains that define the rock in the target volume. The dominant rock type in rock domain RFM029 is a medium-grained metagranite (74% of the domain volume). Subordinate rock types are pegmatitic granite or pegmatite (13%), fine- to medium-grained metagranitoid (5%) and amphibolite and other minor mafic to intermediate rocks (5%). With the exception of amphibolite that contains little or no quartz, the dominant and subordinate rock types have high quartz content (c. 20–50%).

Rock domain RFM045 is surrounded by rock domain RFM029 in the target volume (Figure 4-7) and has a constricted rod-like geometry that plunges moderately to steeply to the south-east (Figure 4-8). Aplitic metagranite and medium-grained metagranite similar to the dominant rock type in domain RFM029 are the dominant rock types (67%) in this domain. More subordinate rock types include pegmatite, and pegmatitic granite (14%), fine- to medium-grained metagranitoid (9%) and amphibolite and other minor mafic to intermediate rocks (7%).

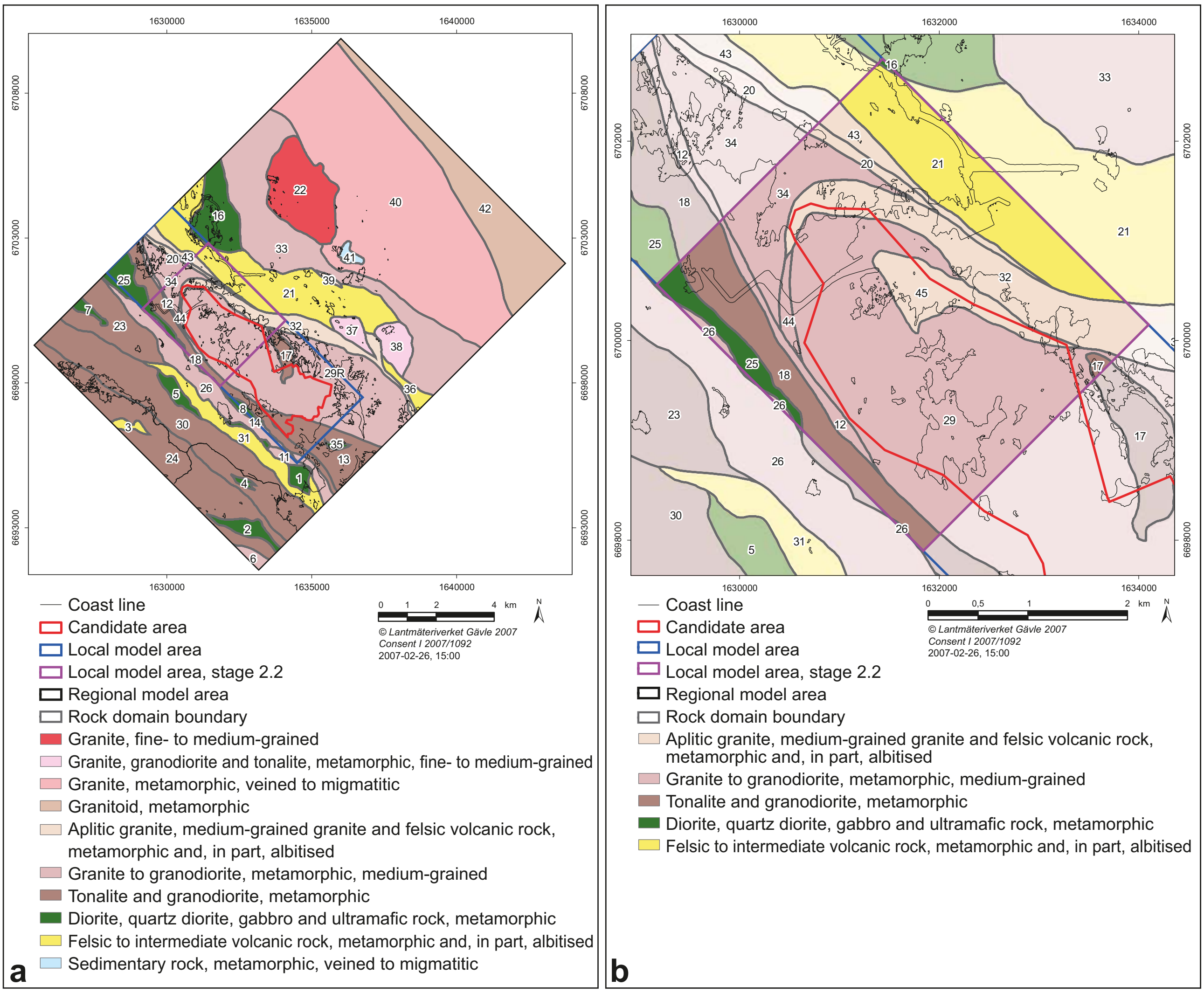


Figure 4-7. *Rock domains included in the two dimensional models at the ground surface. a) Model inside the regional model area. b) Model inside the local model area (darker colours). The different colours represent the dominant rock type in each domain (Figure 5-24 in the* ***Site description Forsmark****).*

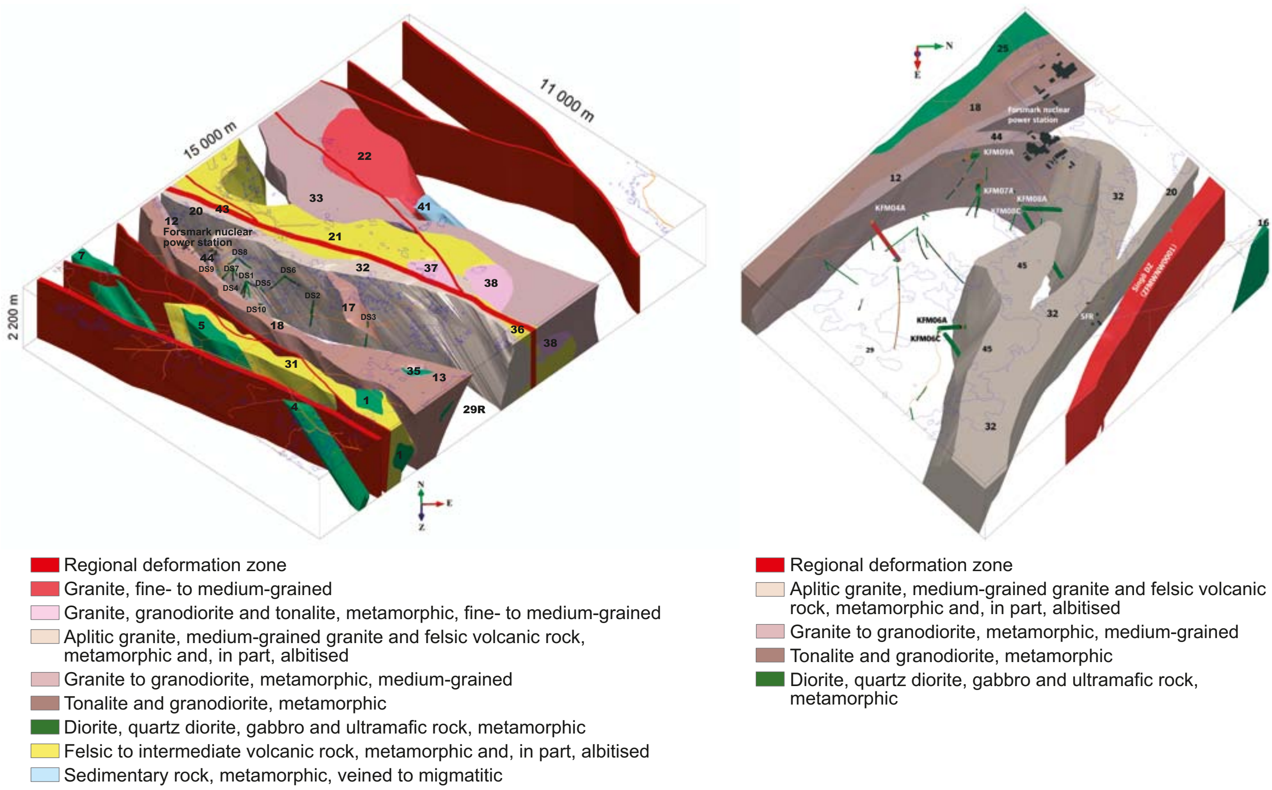


Figure 4-8. *Three dimensional model for rock domains (numbered) and regional deformation zones (red colour). Several domains, including RFM029, are unshaded in order to display the structural style at the Forsmark site and in the tectonic lens. The dominant rock type in each domain is illustrated with the help of different colours (see legend). Left: Regional model showing the modelled, south-eastern elongation of several domains (Figure 4-15 in /Stephens et al. 2007/). Right: Rock domains in the target volume in the north-western part of the candidate area, viewed to the west from approximately the position of SFR (Figure 4-6 in /Stephens et al. 2007/).*

Rock domains outside the target volume

Rock domains outside the tectonic lens and target volume dip steeply towards the south-west, following the trend of the coastal deformation belt (Figure 4-8). They are dominated by different types of granitoid, predominantly felsic volcanic rocks and quartz-poor or quartz-deficient diorite to gabbro. More inhomogeneous bedrock is conspicuous in domains RFM018 and RFM021, on both sides of the tectonic lens (Figure 4-7).

Confidence

Confidence in both the geometry and properties of rock domains within and immediately around the target volume is high down to a depth of 1,000 m, whereas significant uncertainties remain concerning the character and geometry of rock domains outside the target volume, e.g. in the sea area.

4.3.2 Mineral resources

The ore potential in the coastal area in northern Uppland is correlated to the rock types and their characteristics. An assessment of the ore potential came to the conclusion that there is no potential for metallic and industrial mineral deposits within the candidate area at Forsmark. A potential for iron oxide mineralisation was recognised in an area south-west of the candidate area, predominantly in the felsic to metavolcanic rock, but the mineral deposits are small and have been assessed to be of no current economic value /Lindroos et al. 2004/. Felsic to metavolcanic rock is also dominant in rock domain RFM021, located north and offshore of the candidate area (Figure 4-7). There is no documented iron mineralisation in data available from the islands, but since most of this rock domain is located beneath the Baltic Sea from where no mineralogical data exist, the potential for iron oxide mineralisation in rock domain RFM021 cannot be totally excluded.

4.3.3 Thermal properties

The thermal properties, i.e. thermal conductivity and heat capacity, of the rock are closely related to the lithology, since these properties depend on the mineral composition. The thermal conductivity of the rock has been assessed from direct measurements and by calculations based on mineral composition from modal analyses. Relationships between density and thermal conductivity determined from site-specific data support the use of borehole density logging data for modelling the spatial correlation of thermal conductivity within a rock type. These relationships have been shown to be consistent with the results of theoretical calculations of density and thermal conductivity based on the mineralogy of different rock types. The heat capacity has been determined from calorimetric measurements and also indirectly from measurements of thermal conductivity and diffusivity.

Thermal conductivity, heat capacity and temperature

The rock types in rock domain RFM029 have typically high quartz content (c. 20–50%), which implies high values of thermal conductivity. Measurements at the cm-scale show values in the range 3.2 to 4.0 W/(m·K) for the medium-grained metagranite, the dominant rock type in rock domain RFM029. The altered granitic rocks that dominate in rock domain RFM045 also have high thermal conductivity, with measured values in the range 3.6 to 4.0 W/(m·K). However, subordinate rock types in these rock domains yield significantly lower values, e.g. amphibolite in the range 2.2 to 2.5 W/(m·K). Amphibolite occurs as narrow, dyke-like tabular bodies and irregular inclusions that are elongate in the direction of the mineral stretching lineation. Although some bodies are more than a few metres in thickness, most are inferred to be minor rock occurrences, i.e. thin geological entities. Measured and calculated conductivities also indicate that rocks affected by the alteration referred to as oxidation have higher thermal conductivity than their unaltered equivalents.

The variability in thermal conductivity within each rock type and between rock types is incorporated into stochastic modelling at a domain level by the use of spatial statistical models of lithology and thermal conductivity. The mean value of the thermal conductivity from the stochastic simulations is c. 3.6 W/(m·K) at the 5 m scale at a temperature of 20°C, both in rock domain RFM029 and RFM045. The impact of low conductivity rock, mainly the subordinate rock type amphibolite, but also the fine- to medium-grained metagranitoid, is particularly conspicuous in RFM045, a feature displayed by the pronounced lower tail in the resulting histogram of thermal conductivity at the 5 m scale (Figure 4-9).

The mean value of the heat capacity assessed from measurements and from simulated thermal conductivity is 2.1 MJ/(m^3·K), both in rock domain RFM029 and RFM045. This value is valid at a temperature of 20°C, but it does not vary much over the temperature range of interest. The mean value of the heat capacity of the dominant rock type metagranite increases by about 29% per 100°C temperature increase. The *in situ* temperature constitutes the initial temperature condition for a repository. The current mean temperature at 500 m depth is estimated to be 11.6°C, based on measurements in 8 boreholes.

Confidence

There is generally high confidence in the modelled distribution of thermal properties, due to the large amount of thermal data for the homogeneous rock mass. The thermal conductivity distribution is more uncertain for rock domain RFM045 than for rock domain RFM029. For domain RFM045, these uncertainties concern both the overall distribution and its lower tail, and are related to uncertainties associated with the output of the geological simulations, in particular, the proportions of rock types and the spatial and size distribution of amphibolite. Although the geological simulations performed have managed to model much of the heterogeneity observed in the boreholes, it is still somewhat unclear to what extent the borehole information is representative of the geology in domain RFM045. The reason for this is the small number of boreholes combined with the more heterogeneous distribution of amphibolite in this rock domain.

4.3.4 Strength and other mechanical properties of intact rock

Lithology also directly affects the thermal expansion, mechanical strength and deformation properties of the unfractured (intact) rock. The mechanical strength and deformation properties of the rock are evaluated from results of measurements on samples from the dominant rock types in RFM029 and RFM045, and also from the subordinate rock type pegmatite.

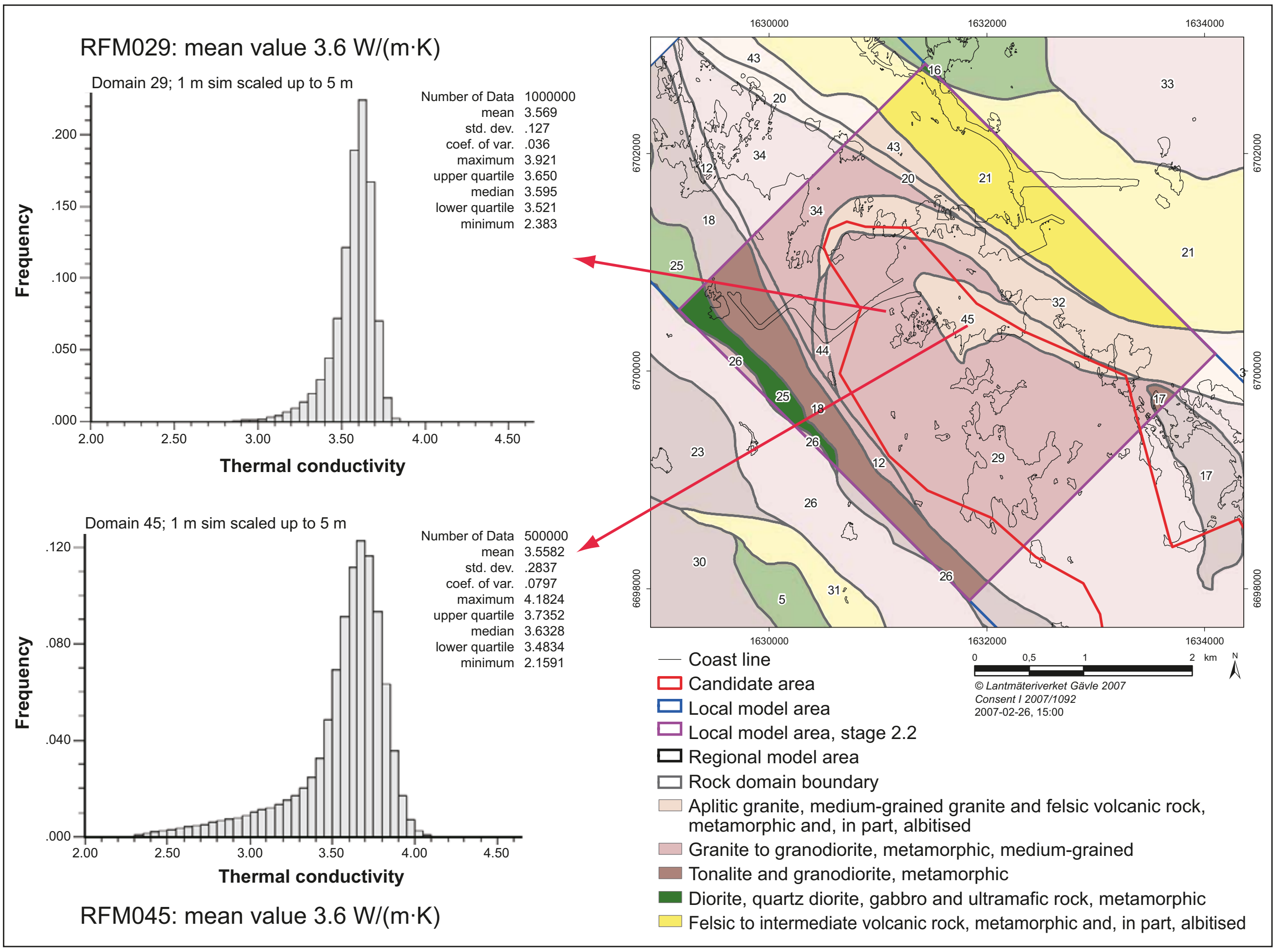


***Figure 4-9.** Modelled thermal conductivity at the 5 m scale. The lower tail of the distribution is mainly due to the subordinate rock type amphibolite (Figure 11-10 in the **Site description Forsmark**).*

Thermal expansion

The mean value of the measured thermal expansion coefficient for the main metagranitoid rock types within the target volume varies between $7.5 \cdot 10^{-6}$ and $7.8 \cdot 10^{-6}$ m/(m·K). The mean of the measurements for the dominant medium-grained metagranite in RFM029 is $7.7 \cdot 10^{-6}$ m/(m·K), and the corresponding value for the altered metagranite, which dominates in RFM045, is $7.5 \cdot 10^{-6}$ m/(m·K). Domain modelling has not been performed, but the small differences in measured values suggest a mean coefficient of thermal expansion of $7–8 \cdot 10^{-6}$ m/(m·K) for the different rock domains.

Strength and deformation modulus

The uniaxial compressive strength (UCS) and Young's Modulus (E) of the intact rock show that the rock types in rock domains RFM029 and RFM045 are strong (UCS > 200 MPa) and stiff (E > 70 GPa). The results for samples taken inside or in the vicinity of deformation zones are in the same range as the results for samples taken in the host rock outside deformation zones.

Confidence

There is generally a high confidence in the strength and deformation properties of the dominant rock types in rock domains RFM029 and RFM045. Some uncertainties remain in the uniaxial compressive strength of the subordinate rock types amphibolite and fine- to medium-grained metagranitoid in these rock domains. However, the proportions of these rock types in rock domains RFM029 and RFM045 inside the target volume are small.

4.4 Deformation zones, fracture domains and fractures

Deformation zones and fractures are themselves important characteristics of the site as they affect the possible location of the repository, the mechanical stability of the rock and the groundwater flow. Furthermore, the deformation history and the geometry of the deformation zones affect the rock stress distribution and thereby also the properties of fractures in the volume.

4.4.1 Formation and reactivation through geological time

Four sets of deformation zones have been identified with high confidence at the Forsmark site (Figure 4-10). Vertical and steeply, SW-dipping zones with subsets referred to as WNW and NW show complex, ductile and brittle deformation. Regional zones longer than 10 km that occur outside the candidate volume (e.g. the Forsmark, Singö and Eckarfjärden deformation zones) are restricted to this set (Figure 4-6). The deformation zones in the remaining three sets only display brittle deformation and can be referred to as fracture zones. Vertical and steeply dipping fracture zones with sub-sets referred to as ENE (NE) and NNE transect the tectonic lens and occur frequently inside the target volume Figure 4-10. These zones formed in the brittle regime and are dominated by sealed fractures and sealed fracture networks. Gently dipping fracture zones occur more frequently in the south-eastern part of the candidate area, i.e. outside the target volume. Relative to the other three sets, there is an increased frequency of open fractures, including crush zones, along the gently dipping set. The fourth set consists of vertical and steeply dipping fracture zones referred to as NNW that are dominated again by sealed fractures. On the basis of their low frequency of occurrence, these are judged to be of lower significance relative to the other three sets of zones at Forsmark.

Low-temperature geochronological data, dating of fracture minerals, kinematic data and consideration of deformation in a regional perspective have been used to establish a conceptual model for the formation and reactivation of deformation zones in the context of changes in the stress regime from the later part of the Svecokarelian orogeny, c. 1.85 to 1.75 Ga, until the Quaternary. This conceptual model suggests that the different sets and sub-sets of deformation zones in the Forsmark area had formed and had already been reactivated during Proterozoic time, in connection with several tectonic events prior to 900 Ma (Figure 4-11). The termination pattern of surface lineaments and the occurrence of ductile deformation indicate that the steeply dipping zones referred to as WNW and

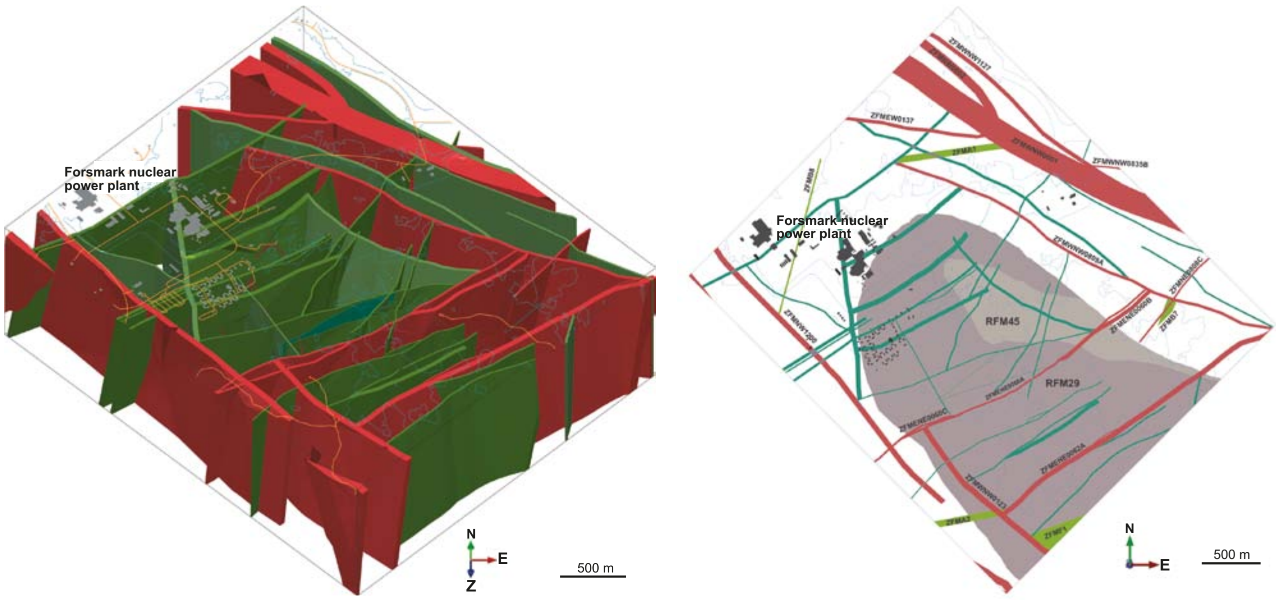


Figure 4-10. Three dimensional model that shows all the vertical and steeply dipping zones in the local model volume (Figure 4-3) (a) and a two dimensional horizontal surface at –500 m elevation in the local model volume including all zones (b) (Figures 5-12 and 5-13 in /Stephens et al. 2007/). Zones marked in red have a trace length at the surface longer than 3,000 m and zones marked in green (a) or blue-green (b) are shorter than 3,000 m in length. Zones marked in pale green (b) are gently dipping.

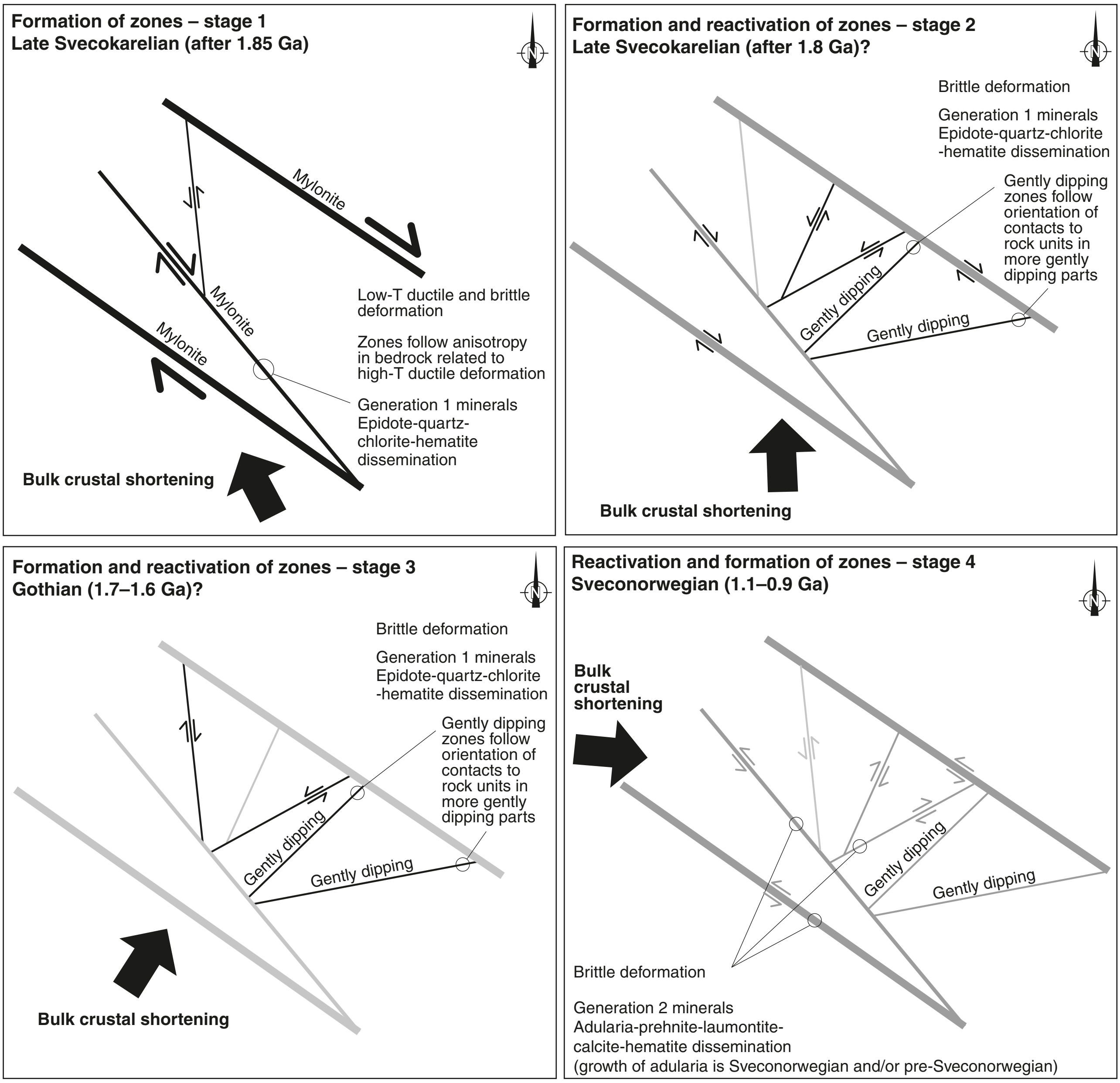


Figure 4-11. Two-dimensional cartoons illustrating the regional scale geodynamics during the formation and reactivation of the different sets of deformation zones at the Forsmark site. This includes late Svecokarelian, low-T ductile and brittle deformation (stage 1), late Svecokarelian brittle deformation (stage 2), Gothian brittle deformation (stage 3), and a major phase of brittle reactivation during the Sveconorwegian orogeny (stage 4). Formation of fractures and fracture zones during stage 4 cannot be ruled out. The different colour shadings along the zones indicate an inferred variable degree of response (strongest is black, intermediate is grey, weakest is pale grey) in each tectonic regime (Figure 5-26 in the ***Site description Forsmark****).*

NW form the oldest discrete structures at the site. It is inferred that they formed in response to bulk crustal shortening in a NW-SE to N-S direction, during the later part of the Svecokarelian orogeny. The size of several of these zones (e.g. Forsmark, Singö and Eckarfjärden) confirms that they comprise the master set. The formation of some steeply dipping NNW structures is also inferred to have occurred at this stage in the tectonic evolution. Displacement along steep NW zones as well as along steep ENE and NNE zones is inferred to be related to approximately N-S compression during the latest part of the Svecokarelian orogeny. Conjugate relationships between steeply dipping ENE and NNW structures are related to approximately NE-SW compression and the younger Gothian tectonic event (1.7–1.6 Ga). It is suggested that the compressive deformation along the gently dipping zones occurred during these tectonic episodes. In such a conceptual model, it can be expected that the gently dipping zones both terminate against the steeply dipping ENE and NNE structures and are displaced by them. The seismic reflection data provide some support to both these statements. Data also demonstrate that the bedrock at Forsmark has been strongly affected by the Sveconorwegian orogenic event (1.1–0.9 Ga).

As the effects of tectonic activity, for the most part, waned, the effects of loading and unloading increased in significance. The latter occurred in connection with the deposition of sedimentary rocks and the subsequent erosion of these rocks with exhumation of the crystalline bedrock. In this respect, the formation of sedimentary basins during the time interval c. 1.5 to 1.3 Ga and after 900 Ma, an episode of glaciation during the latest part of the Precambrian around c. 650 Ma, the development of a passive continental margin during the Early Palaeozoic, and numerous glaciations during the Quaternary period are examples of loading events during the geological development of central Sweden. It is also suggested that sedimentary loading is a process that, besides tectonic events, gave rise to the build-up of high rock stress in the bedrock. By contrast, unloading resulted in reactivation, especially of gently dipping structures, in the form of extensional failure and the development of dilatational joints. New fractures that are oriented sub-parallel to the topographic surface at the time of unloading and lack alteration associated with hydrothermal alteration, i.e. sheet joints, would also have formed (see Section 4.10 and Figure 4-24). These features occurred in response to a release of stress in the bedrock. They are most conspicuous close to the surface interface, where the differential stress ($\sigma_1 - \sigma_3$) at the time of unloading was high, and, especially, in the vicinity of ancient gently dipping zones.

The Baltic Shield is currently affected by two large-scale deformation processes; plate tectonics and glacial isostatic adjustment due to retreat of the most recent Fennoscandian ice sheet, see further the **Geosphere process report**. The plate tectonic component includes ridge push from the west due to seafloor spreading at the Mid-Atlantic Ridge and compression from the south due to the Eurasia-Africa plate collision. The current tectonic stress field in the Forsmark region is dictated by the Mid-Atlantic Ridge push (see also Section 4.5.2).

4.4.2 Deterministic deformation zones

In the geological model, structures that are 1,000 m or longer are included in the deterministic deformation zone model, whereas structures shorter than 1,000 m are described in statistical terms in the geological DFN (discrete fracture network) model (see Section 4.4.3). The deterministic deformation zone model builds on the integration of the understanding of the deformational history in the region, magnetic lineament and seismic reflection data, as well as fracture orientation, fracture mineralogical and alteration data, especially from the cored boreholes. Low magnetic lineaments derived from both high-resolution ground magnetic data and airborne helicopter data inside and immediately around the target area are the main surface data that support the occurrence of steeply dipping deformation zones. Geometrical constraints on the occurrence of gently dipping fracture zones are provided by surface reflection seismic data and borehole seismic data (VSP). Reflectors that dip to the SSE and SE are prominent in the candidate volume, with a much stronger concentration in the upper 2 km of the bedrock in the south-eastern part relative to that observed in the north-western part, i.e. the target volume (Figure 4-12). Borehole data show that several of the gently dipping fracture zones in the south-eastern part of the candidate volume occur along or close to the contact between subordinate rock types, in particular amphibolite, and the dominant host rock metagranite. Since the amphibolite contacts follow the orientation of the tectonic foliation, it is proposed that the more frequent occurrence of gently dipping zones in the south-eastern part is related to the gentler, south-east dip of the amphibolite, the tectonic foliation and the mineral stretching lineation in this part of the candidate volume.

The local model, which contains the target volume (Figure 4-3), includes 60 deterministically modelled deformation zones, and the majority of these zones (> 60%) are judged to have a high confidence of existence (Figure 4-10). Only two steeply dipping zones with a surface trace length longer than 3,000 m intersect the target volume (zones ENE0060A and ENE0062A with their attached branches) and a further twenty-two steeply dipping zones, which either show a trace length at the ground surface between 1,000 and 3,000 m or form minor splays or attached branches to such zones, are present at 400 to 600 m depth inside the repository volume. Five gently dipping zones, including zones A2 and F1, are also present at 400 to 600 m depth inside or immediately above the repository volume.

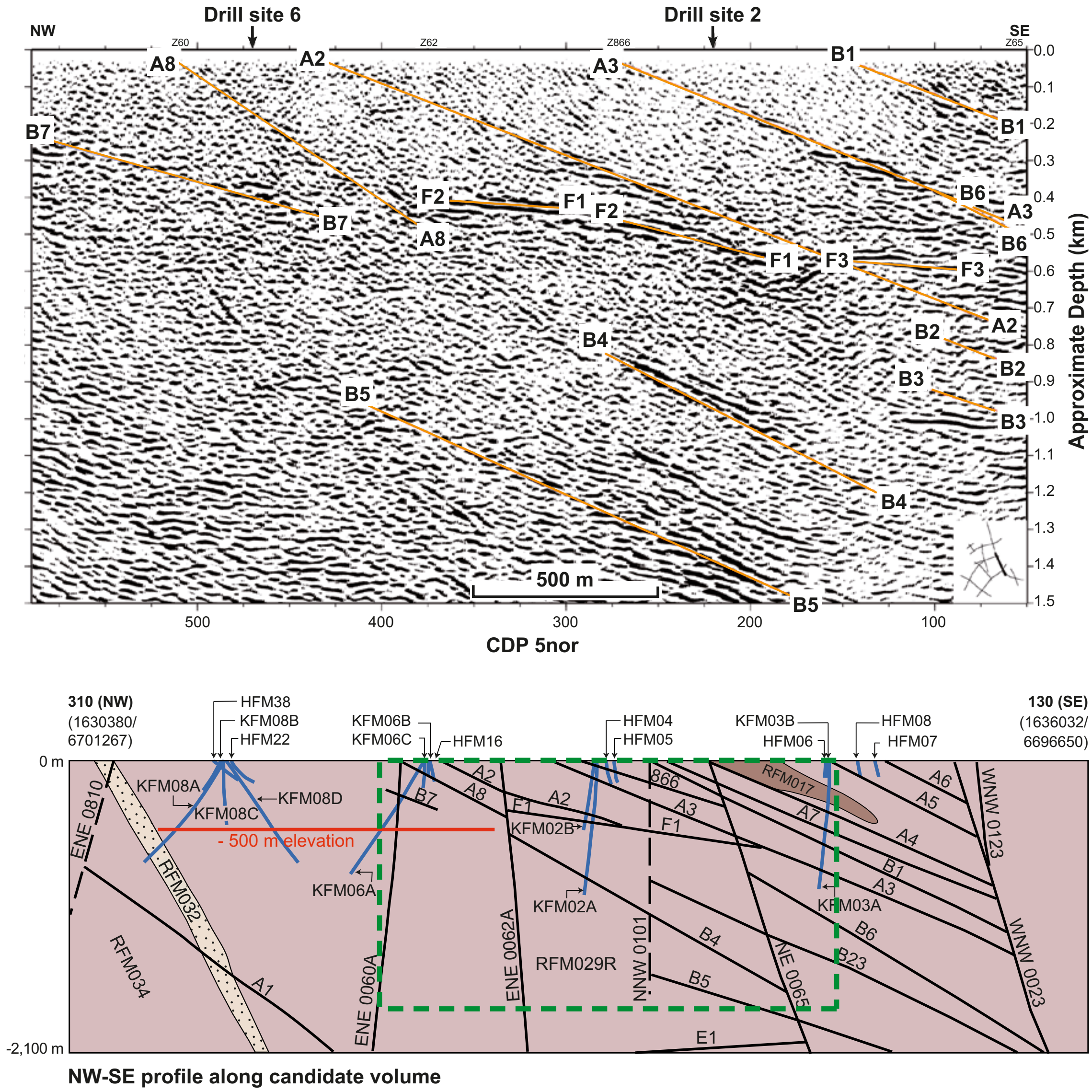


Figure 4-12. *Confidence in the gently dipping fracture zones is enhanced by combining reflection seismic data with borehole data. The upper insert shows gently dipping seismic reflectors in the south-eastern part of the candidate volume. The lower insert is a NW-SE cross-section through the candidate volume in the structural model showing rock domains and deformation zones. The approximate location of the upper insert is indicated by green dashed lines in the lower insert. The cross-sections are not identical (Figure 11-13 in the* ***Site description Forsmark****).*

The regional model includes 72 deterministically modelled deformation zones of which 29 are present in the local model. The minority of the zones in the regional model (27) are judged to have a high confidence of existence. The remainder lack corroborative geological and geophysical data from boreholes or excavations and are judged to have a medium confidence of existence. As in the local model, vertical and steeply dipping deformation zones (48) dominate over gently dipping zones (24). However, there is an inherent bias in the evaluation of the occurrence of gently dipping zones, since reflection seismic data are not available over the north-eastern half of the regional model volume.

Confidence

Confidence in the occurrence of deterministic deformation zones in the target volume is high and the occurrence of undetected deformation zones longer than 3,000 m is judged unlikely. It is considered that the deterministic model for deformation zones has now attained acceptable stability, in both the local and regional model volumes. The predictability of the occurrence and character of different sets or sub-sets of deformation zones is related to the large-scale bedrock anisotropy that was established over 1.85 billion years ago, when the bedrock was situated at mid-crustal depths and was affected by penetrative, ductile deformation under high-temperature metamorphic conditions.

The principal remaining uncertainty in the deterministic deformation zone model concerns the size of the gently dipping fracture zones. However, it is judged that this uncertainty is sufficiently bounded, since the approach for termination of these zones has been to extend them to the nearest steeply dipping zone. Furthermore, this uncertainty is not fundamental to the conceptual understanding of the site. Another remaining uncertainty concerns the orientation and size of the possible deformation zones that have not been modelled deterministically. Since it has not been possible to link these geological features to low magnetic lineaments or seismic reflectors and since they commonly occur along short borehole intervals, it is judged that they are predominantly minor zones.

4.4.3 Fracture domains, fractures and DFN models

Analyses of fracture data have indicated a large degree of spatial variability in the size, intensity and properties of fractures between different rock domains, and also within rock domain RFM029. Based on a systematic assessment of the variation in the frequency of fractures with depth along each borehole, the bedrock between deterministically modelled deformation zones has been divided into fracture domains. Thus, fracture domains and deterministically modelled deformation zones are mutually exclusive volumes, whereas rock domains contain both fracture domains and deterministically modelled deformation zones.

Fracture domains

Fracture domain FFM01 forms the bedrock at depth in the target volume, north-west of and in the footwall of the gently dipping zone A2 (Figure 4-13). This fracture domain forms the main component in rock domain RFM029. Fracture domain FFM02 comprises the bedrock close to the surface, above fracture domain FFM01, predominantly in the same footwall bedrock segment. Fracture domain FFM02 is located in both rock domains RFM029 and RFM045. The bedrock in FFM01 shows a low frequency of open and partly open fractures, whereas the bedrock in FFM02 is characterised by a complex network of sub-horizontal or gently dipping, open and partly open fractures, which, locally, merge into minor zones. The sub-horizontal or gently dipping fractures are oriented at a large angle to the current, minimum principal (vertical) stress in the bedrock. The rock volume south-east of the target volume, in the hanging wall of zone A2, is defined as fracture domain FFM03. It is situated in rock domains RFM029 and RFM017. Open and partly open fractures in this domain are more evenly distributed from surface down to 1,000 m depth than in the rock north-west of zone A2, and the domain is spatially associated with a high frequency of gently dipping fracture zones containing both open and sealed fractures. A fourth fracture domain, FFM06, is defined in the target volume. In the same manner as fracture domain FFM01, it lies in the footwall of zone A2 and beneath FFM02. It forms the main component in rock domain RFM045. It was distinguished from FFM01 simply on the basis of the widespread occurrence of fine-grained, altered (albitised) granitic rock, with slightly higher contents of quartz compared with unaltered granitic rock.

Borehole data have revealed that the orientation and mineralogy of fractures inside fracture domains FFM01 and FFM06, as well as in FFM03, are similar to that in the adjacent fracture zones. However, this correlation breaks down in domain FFM02, where there is an important contribution from sub-horizontal and gently dipping fractures, but relatively few gently dipping zones. It is inferred that tectonic processes during the geological history of the region have determined the fracturing at depth in fracture domains FFM01 and FFM06, whereas the fracturing in the near-surface bedrock in fracture domain FFM02 is the result of a combination of tectonic processes and processes related to stress release.

Discrete fracture network models

Fractures and minor fracture zones, not covered by the deformation zone model, are handled in a statistical way through discrete fracture network (DFN) models. The geological DFN model captures all fractures (open, partly open and sealed), since the sealed fractures are also assumed to be potential planes of weakness and could possibly be planes where there exist flow channels. However, many of

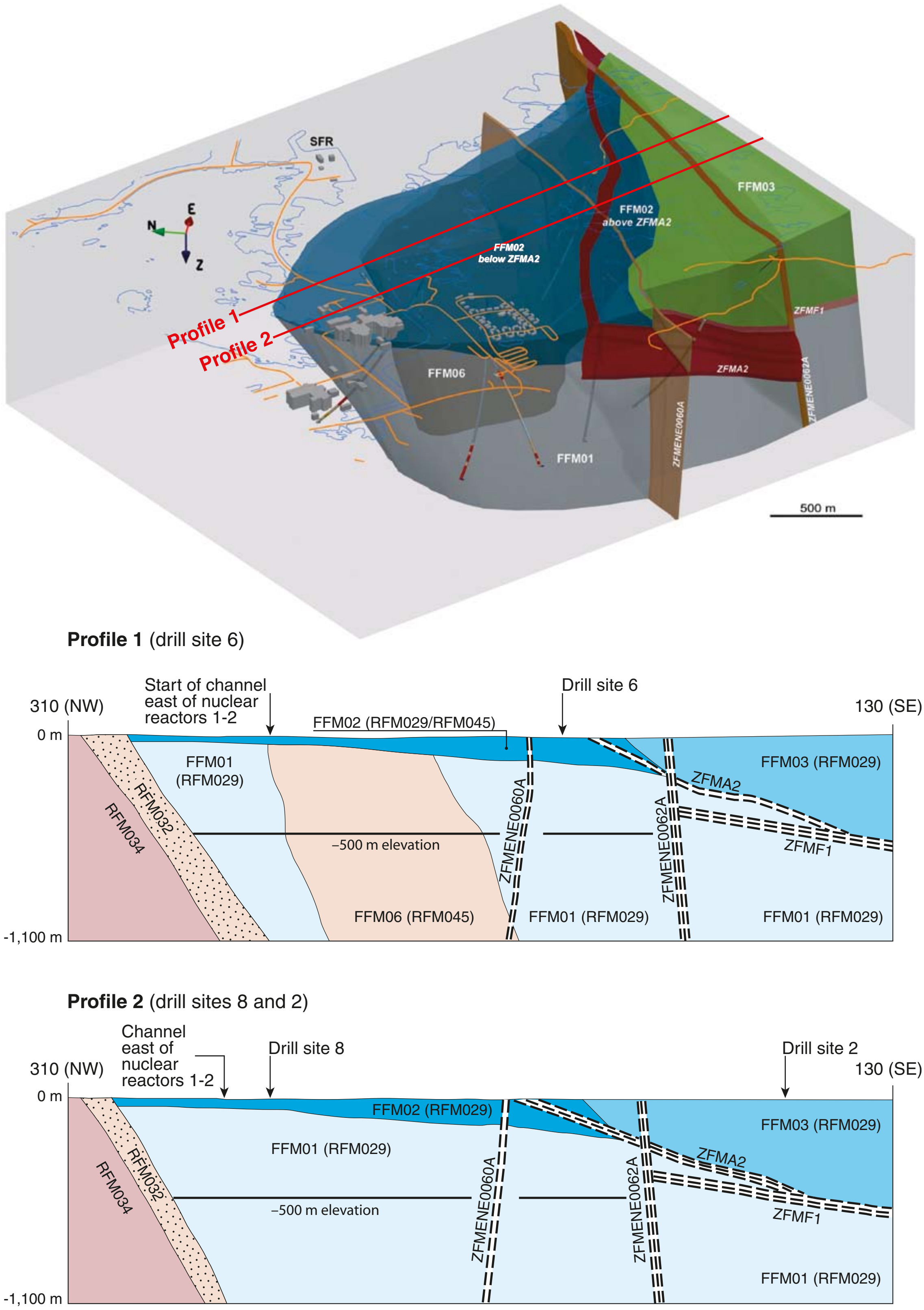


***Figure 4-13.** View to the ENE (upper) and vertical profiles in a NW-SE direction (middle and lower) showing the fracture domains and their relation to the gently dipping deformation zones ZFMA2 and ZFMF1 and the steeply dipping zones ZFMENE0060A and ZFMENE0062A that are longer than 3,000 m (Figure 11-14 in the **Site description Forsmark**).*

the sealed fractures are mechanically and hydrogeologically indistinguishable from the intact rock. The model is presented as a mathematical description of fractures and fracture zones shorter than 1 km, not as a 3D object model or realisation. Thus, the derived model parameters can be used to stochastically model these fractures in various applications.

DFN models are produced for each fracture domain in the target volume, i.e. fracture domains FFM01, FFM02 and FFM06, and also for fracture domain FFM03, based on fracture trace length data observed in outcrop, data derived from the interpretation of magnetic lineaments, data derived from the lengths of the intercepts of the mid-planes of the deterministic deformation zones and fracture intensity data from boreholes. The influence of different geological processes at depth and close to the surface causes difficulties in applying coupled size-intensity geological DFN models, which are based on surface data, to the two fracture domains at repository depth that do not outcrop, FFM01 and FFM06. There are also uncertainties regarding whether the fractures represented by deformation zones and magnetic lineaments are part of the same fracture population represented by fractures in outcrop or measured in boreholes. These difficulties are captured by alternative size-intensity models for the distribution of fractures. Details of these alternative models and their associated uncertainties are described in the **Data report**, Section 6.3.

Confidence

There is high confidence in the occurrence of different fracture characteristics in the bedrock outside deformation zones in the target volume and, consequently, in the division of this bedrock into a more fractured near-surface volume, fracture domain FFM02, above less fractured rock volumes, fracture domains FFM01 and FFM06. This division is also supported by hydrogeological and hydrogeochemical data (see Sections 4.6 and 4.8), although the position of the boundary between FFM02 and the fracture domains beneath is still associated with uncertainties. However, the most important remaining uncertainty concerns the DFN model for prediction of the size-intensity relationship of fracturing in the potential repository volume, i.e. fracture domains FFM01 and FFM06. The main reason for this is the lack of data on fracture sizes in these sub-surface domains. Direct data on fracture sizes in these domains can only be obtained from underground mapping, i.e. when the repository excavation has reached relevant depths. This uncertainty as well as the question of tectonic continuity at different size scales has been addressed with the help of alternative DFN models. These alternatives cover a broad range and the uncertainties are judged to be bound by them.

4.4.4 Fracture mineralogy

Detailed studies of fracture mineralogy and wall rock alteration have provided information on the character and frequency of fracture minerals at Forsmark. Calcite and chlorite, partly associated with corrensite, are by far the most common minerals. Other common minerals are laumontite, adularia, quartz, albite and hematite, whereas prehnite, pyrite, clay minerals and epidote are less common. Rare occurrences of e.g. asphaltite and goethite are also observed.

The older generations of minerals, including epidote, which formed prior to 1.1 Ga, and adularia, which formed around 1.1 Ga or is older, show no depth dependence. In a similar manner, the wall-rock alteration associated with these mineral generations and referred to as oxidation also shows no depth dependence. These features are consistent with the conclusion that these minerals and the associated alteration formed a long time ago when this part of the bedrock was situated at considerably greater depths. The younger mineral asphaltite, which probably formed between c. 500 and 250 million years ago, occurs almost entirely in the upper part of the bedrock along open fractures. This is also the case for clay minerals and goethite that belong to the youngest generation of minerals, except for some occurrences at greater depths along individual fractures in deformation zones. Since the current bedrock surface is more or less the same surface as the sub-Cambrian peneplain, this distribution of fracture minerals with depth suggests that the near-surface bedrock, i.e. including fracture domain FFM02 in the target volume, has been affected by near-surface processes inferred to be related to loading and unloading cycles during the last c. 500 million years.

Uranium-rich phases are present in some of the fracture coatings, but only one small grain of pitch-blende has so far been detected in a single fracture coating in one of the deterministically modelled gently dipping fracture zones (ZFMB1), located outside the target volume. The origin of these uranium-rich phases is largely unknown, but it can be concluded that uranium has been circulating in the fracture system during different periods throughout the geological history of the site. There is also evidence of redistribution and deposition of uranium irregularly along permeable structures during the Proterozoic (see Section 4.8).

Fractures without mineral coating or filling have been observed in the near-surface realm in the bedrock, but are also present at greater depths. The significance and origin of these fractures have been the focus of a specific study /Claesson Liljedahl et al. 2010/. A subset of fractures from different depths, rock domains and located outside deformation zones were selected for a more detailed investigation. This revealed that the majority of the investigated fractures contain fracture minerals, but that fractures without minerals (5.7% of the inspected fractures) are present in the bedrock above 300 metres depth. Most of these fractures without minerals have been recorded as water bearing and are also sub-parallel to nearby fractures. Some of these nearby fractures are open, but their size is very small and they are often not connected all the way through the drill core, leaving the drill core unbroken.

There is currently no unambiguous process explaining the formation of these fractures without minerals. One hypothetical possibility is that they were formed during opening of fractures or that they represent parts of fracture surfaces that were opened up recently, and therefore have not had enough time to allow for mineral precipitation to occur on the fracture surface. Another possibility is that the non-mineralised fractures represent the fronts of larger fracture surfaces where mineral precipitation may have been prevented by insignificant flow of saturated fluids along the peripheral parts of the fractures. It is not possible to determine the formation age of these fractures. However, given that some of the fracture samples with similar characteristic very small fractures contain hydroxyapophyllite, typical for Palaeozoic generation 3 fracture minerals /Sandström et al. 2008, 2009/, it is possible that the non-mineralised fractures represent reactivation of Palaeozoic fractures. Given the fresh appearance and lack of fracture minerals, it is unlikely that these fractures have been water conductive during the main events of fluid migration in the area, which caused abundant precipitation of fracture minerals. Thus, these fractures have most likely not been open before 277 Ma. Given the sub-horizontal orientation of the identified fractures, it is likely that they have been reactivated due to denudation associated with sediment erosion and/or post-glacial rebound.

4.4.5 Mechanical properties of deformation zones and fractures

Evaluation of the results from laboratory testing of the mechanical properties of fractures has shown that the deformability and strength properties of open discrete fractures are similar for different fracture sets. Furthermore, the properties of open fractures in fracture domain FFM01 and in deformation zones are quite similar. Empirical and numerical analyses of the data show that the deformation modulus and strength properties of the rock mass (including fractures) in fracture domains FFM01 and FFM06 are almost equal and representative of a stiff and strong rock (Figure 4-14).

Confidence

The confidence in the derived mechanical properties of the rock mass is in general high, due to the large amount of data in support of the model and the small changes in values caused by the addition of new data during the various modelling stages. In addition, the results are consistent with the understanding of the geology at the site. The largest remaining uncertainty concerns the large-scale mechanical properties of fractures, since the model is based on test results on small samples.

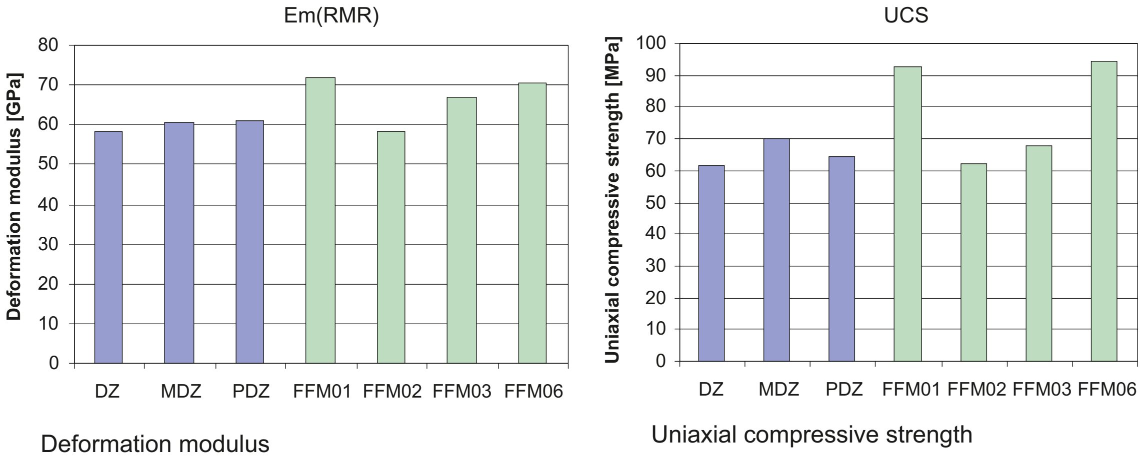


Figure 4-14. *Mean values of deformation modulus and strength properties of the rock mass in fracture domains and deformation zones as evaluated by an empirical approach (Figure 11-16 in* ***Site description Forsmark****). DZ = deformation zone, MDZ = minor deformation zone (shorter than 1,000 m), PDZ = possible deformation zone.*

4.5 Rock stress

4.5.1 Stress evolution

The stress evolution is closely related to the deformational history of the site. According to the conceptual understanding (Section 4.4), the different sets and sub-sets of deformation zones at the site formed and had already been reactivated more than 900 million years ago in response to stress conditions affected by different tectonic events along active continental margins. This, in combination with loading and unloading cycles connected with the burial and denudation of sedimentary rocks and glaciation and deglaciation, has most likely played an important role in the evolution of the stress in the bedrock in the Forsmark area, and specifically the bedrock inside the target volume.

The character of the bedrock close to the surface in the target volume, FFM02, is related to the unloading of the rock and the release of *in situ* stress. Reactivation of ancient fractures and even formation of new fractures (sheet joints) is apparent, at least close to the surface, during the Quaternary. The rock in the south-eastern part of the candidate area, FFM03, is spatially associated with a high frequency of gently dipping fracture zones containing both open and sealed fractures. These structural features are consistent with a general stress-released region.

4.5.2 Stress model

The assessment of the *in situ* stress state at the Forsmark site is based on both direct measurements and indirect observations. Data from direct measurements by overcoring, hydraulic fracturing and hydraulic tests on pre-existing fractures are available from a number of boreholes in fracture domains FFM01, FFM02 and FFM03. Indirect observations from boreholes in the same fracture domains include observations of core disking and borehole breakouts in c. 10 km of borehole walls down to depths of 1,000 m. In addition, estimates of the micro-crack porosity in samples from boreholes were used as indirect indicators in the evaluation of stress. Direct measurements as well as indirect observations all indicate a general orientation of the maximum horizontal stress in the range of N120° to 150°. This major principal stress orientation is consistent with the orientation of regional compression derived from seismic studies, as well as with the overall trend for NW Europe due to the Mid-Atlantic Ridge push.

The stress model for the target volume (Figure 4-15) is developed on the basis of data from overcoring measurements and evaluations of indirect observations, combined with the understanding of the geological conditions at the site and an evaluation of other external influencing factors such as topography, glacial rebound and crustal thickness. As shown by data and supported by findings from regional seismicity studies and the understanding of the deformational history at the site, the magnitudes of both the maximum and minimum horizontal stresses are greater than the vertical stress. The increase in the horizontal stress magnitudes with depth in fracture domain FFM01 appears to correlate with the decrease in frequency of open fractures with depth and a corresponding increase in the rock mass stiffness. The stress model implies a most likely value of the maximum horizontal stress of c. 41 MPa and c. 23 MPa for the minimum horizontal stress at 500 m depth in fracture domain FFM01 in the target volume. The estimated vertical stress at this depth in fracture domain FFM01 is c. 13 MPa. No stress measurements have been carried out in FFM06, but the *in situ* stress state is expected to be similar to that in FFM01, since it consists of a rock mass with similar stiffness properties and is located next to FFM01, below the gently dipping deformation zone A2.

Based on results of numerical modelling, it is inferred that the steeply dipping deformation zones in the target volume cause only small perturbations in the stress field, whereas the effect of the gently dipping zones A2 and F1 is more pronounced, with significantly higher stress magnitude below relative to that above these zones. The results further show that release of stress in the hanging wall has reached deeper levels south-east of zone A2 relative to that below and north-west of this zone, i.e. in the target volume (Figure 4-15). This is supported by results from overcoring measurements that show an average horizontal stress magnitude approximately 50% lower in the rock above zone A2 (fracture domain FFM03) compared with the horizontal stress magnitude measured at comparable depths in the target volume. Furthermore, these findings are consistent with the interpretation that the occurrence of the gently dipping zones favoured stress release in the south-eastern part of the candidate volume.

Confidence

The confidence in the orientation of the rock stresses is high due to the consistency in results from the different measuring methods and indirect observations. It also agrees with regional seismic studies. The confidence in the vertical stress magnitude is also high, since measured values and theoretical values based on the weight of the overlying rock cover are in concordance. Uncertainty remains in the magnitude of the horizontal stresses at repository depth. However, an upper bound solution was used to constrain these stress magnitudes and hence it is judged that the uncertainty is sufficiently well constrained.

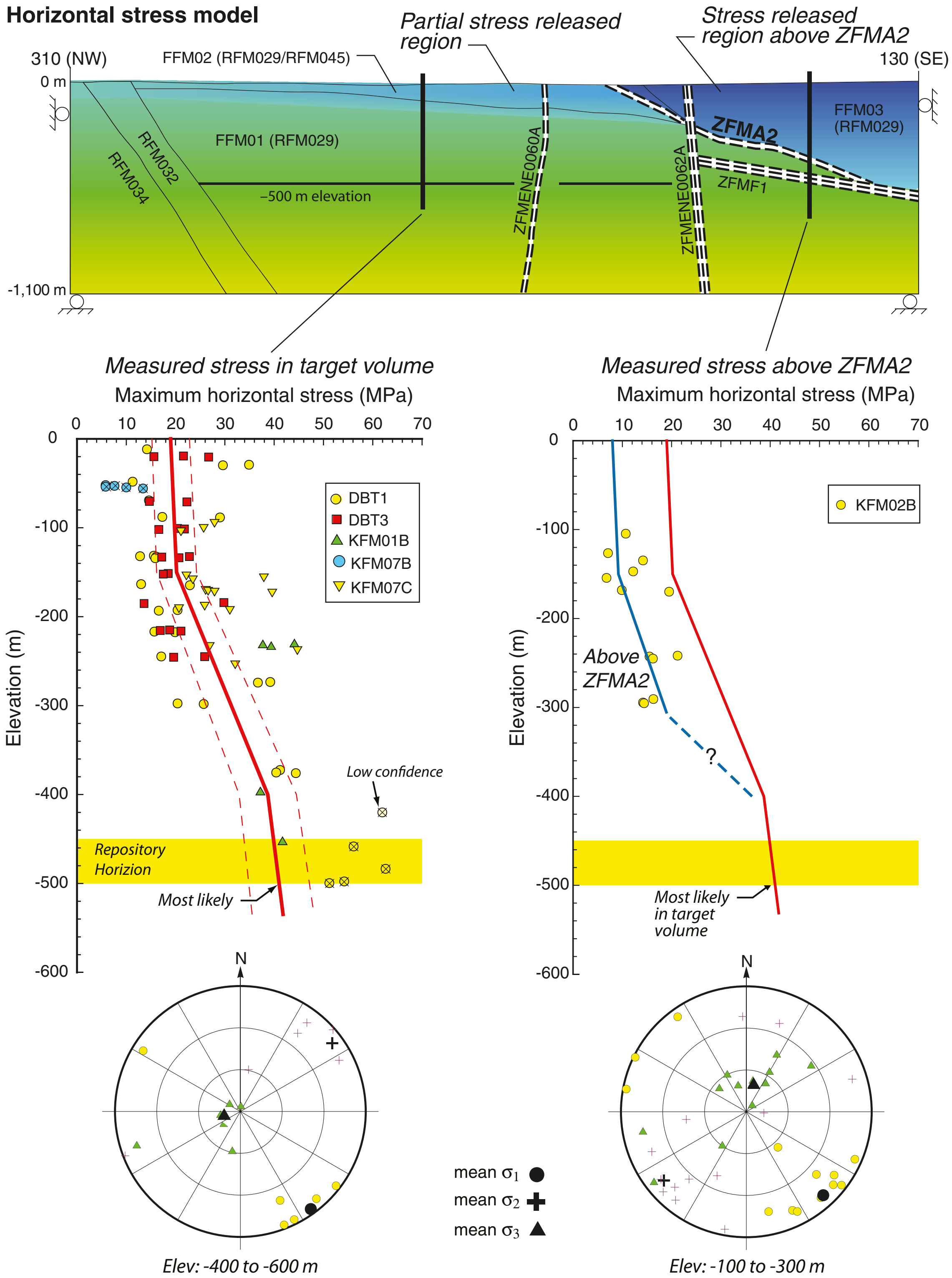


Figure 4-15. *Comparison of the measured maximum horizontal stress magnitudes in the target volume with those measured above the gently dipping deformation zone A2 (central insert). The change in slope in the target volume at c. 150 m depth corresponds approximately to the boundary between fracture domains FFM02 and FFM01. An illustration of the distribution of maximum horizontal stresses from numerical modelling is also shown (upper insert), with the blue shades representing reduced stress magnitudes caused by stress release above zone A2 and in fracture domain FFM02 compared with the "normal" stress magnitudes shaded green. The orientations of the measured principal stresses are shown in two lower hemisphere stereonets (lower insert). The principal stresses show generally consistent orientations regardless of depth and spatial location (Figure 11-18 in the* ***Site description Forsmark****).*

4.6 Bedrock hydraulic properties

4.6.1 Evolution

Geological data at the site, including the dating of fracture minerals, indicate that a large proportion of the fractures are sealed (c. 75%) and that the majority of the fractures are ancient structures, especially in the deeper parts of the bedrock in the target volume, i.e. fracture domains FFM01 and FFM06. The youngest generation of calcite occurs in fractures and deformation zones that are currently hydraulically conductive and may have precipitated during a long period including the present. Based on these observations and indications, it is conceptually attractive to envisage that the hydraulic properties of fractures can be correlated with both the brittle deformational history that formed and reactivated the fractures, and with the formation of fracture minerals during different periods in the past. In contrast, an analysis of transmissivity data versus normal stress suggests a poor correlation, if any. Most likely, the processes that occurred during the past 1.85 billion years overshadow any coupling to the current stress field, which has existed for approximately the past 12 million years. Notwithstanding these considerations, the question of correlation may be scale dependent.

4.6.2 Hydraulic properties of deformation zones and fracture domains

Deterministically modelled deformation zones

Detailed information on the position of deterministically modelled deformation zones and water-conductive fractures in between deformation zones combined with results of high-resolution inflow measurements in 22 core-drilled boreholes and pumping tests in 32 percussion-drilled boreholes provide the basis for the assignment of hydraulic properties of deformation zones and fracture domains at the Forsmark site. These data constrain locations of connected flowing fractures with high certainty and values of the integrated transmissivity for the connected fractures down to c. 10^{-9} m^2/s. Results from single-hole injection tests and multiple-hole interference tests complement the data base.

Analyses of the hydraulic data have revealed that all deterministically modelled deformation zones, regardless of orientation, are characterised by a substantial decrease in transmissivity with depth, with a contrast of c. 20,000 times over the uppermost 1,000 m of the bedrock (Figure 4-16 upper insert). The lateral heterogeneity at each depth is also substantial, but more irregular, suggesting a channelled flow field within the planes of the deformation zones. Furthermore, the data show that the gently dipping deformation zones, which predominantly occur in the south-eastern part of the candidate volume, are the most transmissive at each depth (Figure 4-16 upper insert). The steeply dipping deformation zones that strike WNW and NW and border the candidate area form structures with a second order of importance as far as transmissivity is concerned. These observations provide support for the hypothesis that involves a pronounced hydraulic anisotropy on a regional scale, where the largest transmissivities observed are associated with deformation zones that are oriented at a high angle to the minimum principal stress or sub-parallel to the maximum horizontal stress, respectively.

In the model, the observed vertical and lateral heterogeneity in transmissivity of the zones are both honoured. An exponential decrease in transmissivity with depth is assigned based on the depth trend in the data. The lateral heterogeneity in transmissivity is represented statistically as a lognormal distribution at each depth based on the variability observed in the data. This is further quantified in the **Data report**, Section 6.6.

Fracture domains

The hydraulic characterisation of the rock in between deformation zones supports the division of the bedrock into different fracture domains. In addition, the hydraulic data suggest that the subsurface fracture domain FFM01, located in the target bedrock volume, should be further divided into three depth intervals (Figure 4-16, lower left insert). Below 400 m depth, very few flowing fractures occur, and these are predominantly sub-horizontal fractures with an average spacing of c. 200 m. However, it should be noted that the frequency of flowing fractures below 400 m is so low that it is highly speculative to define an average separation. Above 400 m depth, the frequency of flowing fractures is somewhat higher up to about 200 m depth. Above 200 m depth, there is a significantly higher frequency of flowing fractures, not least fractures with high transmissivities. In FFM03, there are far less boreholes to substantiate an analysis of hydraulic subdomains, but as shown in Figure 4-16 (lower right insert), there is also some depth dependence in this domain.

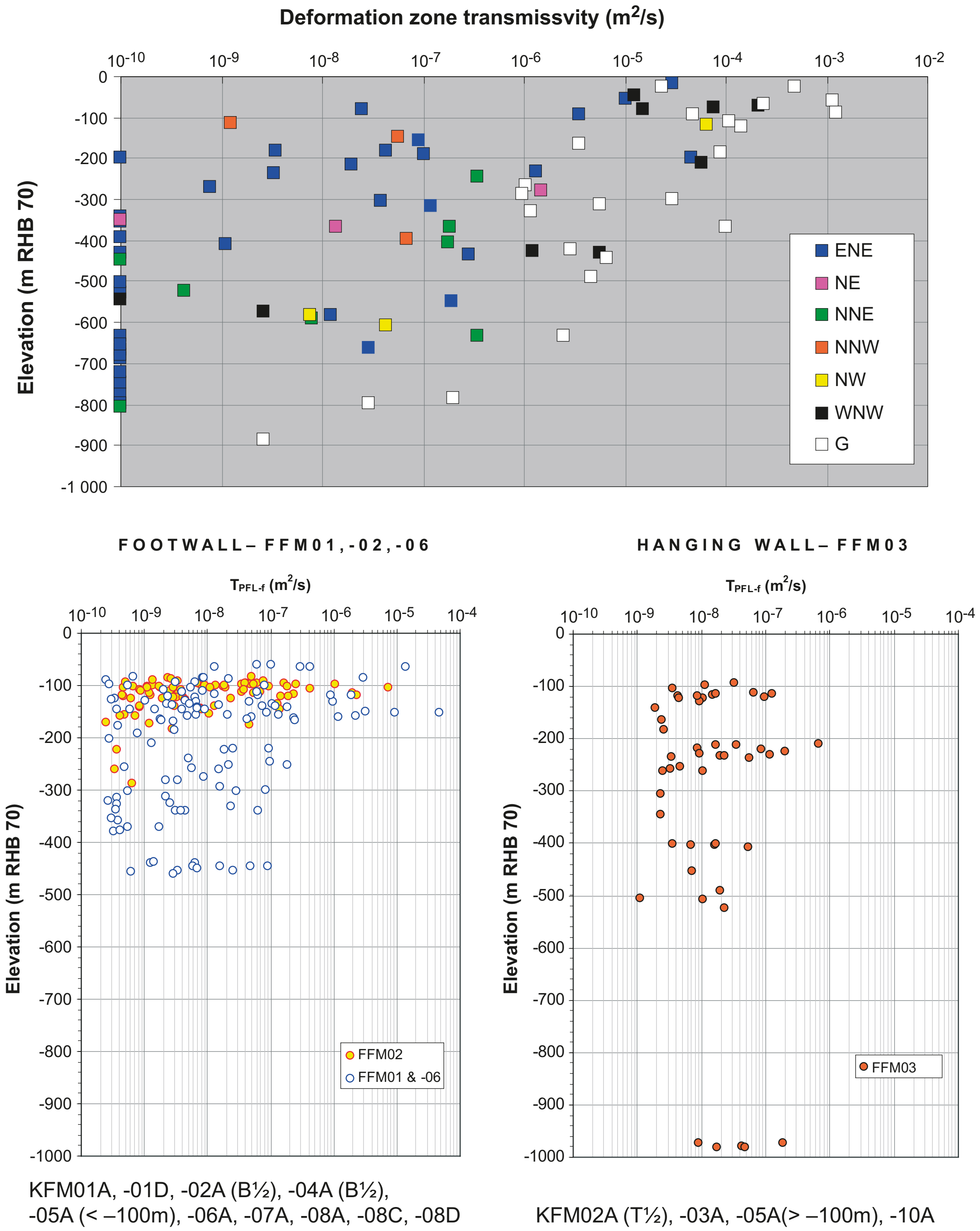


Figure 4-16. *Top: Inferred transmissivities of deformation zones with depth and orientation (G = gently dipping zones). Bottom: Inferred transmissivities of connected open fractures with depth in FFM01, FFM02 and FFM06 (left) and FFM03 (right) (Figure 11-19 in the* ***Site description Forsmark****). Note that the bottom left insert contains data from 10 boreholes whereas the bottom right insert contains data from 4 boreholes only (B½ = bottom half of the borehole, T½ = top half of the borehole).*

The geometric and hydraulic properties of fractures in the rock between the deterministically modelled deformation zones are represented by DFN models calibrated against fracture frequency data and hydraulic data from boreholes. This calibration is made for fracture domains FFM01, FFM02 and FFM03. Fracture domain FFM06 is inferred to have the same properties as fracture domain FFM01. The division into hydraulic subdomains honours the variation in the intensity and size of open fractures with depth inside fracture domain FFM01. Uncertainties in transmissivity of the connected fracture network are handled by alternative assumptions concerning the possible relationships between fracture transmissivity and size (correlated, semi-correlated and uncorrelated), see **Site description Forsmark**, Section 8.5.2 for details. Although it is difficult to establish which of these models may best reflect reality, all three models yield similar ranges of transmissivities for fractures in the size range 10 to 100 m. For this reason, the three models are likely to show similar flow characteristics. The hydraulic DFN models and their uncertainties are further quantified and discussed in the **Data report**, Section 6.6.

Uppermost part of the bedrock

There are three pieces of evidence that indicate a well-connected network of highly transmissive structures in the uppermost c. 150 m of the bedrock in the target volume where fracture domain FFM02 occurs: 1) exceptionally high water yields in the percussion-drilled boreholes (Figure 4-17), 2) the nearly uniform groundwater levels in the uppermost part of the bedrock, and 3) the extensive and rapid transmission of fluid pressure changes during the large-scale interference tests conducted within the target volume. These structures are inferred to be large sub-horizontal fractures, so-called sheet joints, which are known to exist close to the surface (Figure 4-24). The network is found to short circuit the recharge from above as well as the discharge from below. This is hydrogeologically conceptualised as a shallow, anisotropic, bedrock aquifer on top of a thicker segment of bedrock with aquitard type properties (Figure 4-18). Based on the occurrence of high transmissivities, the lateral extent of this bedrock aquifer is envisaged to correspond approximately to the lateral extent of fracture domain FFM02. However, the hydrogeological data indicate that this network of structures probably extends to the north-east as far as the Singö deformation zone.

Figure 4-17. *Two key features of the bedrock in the target area at Forsmark. Left: High water yields are often observed in the uppermost c. 150 m of the bedrock. Right: The large number of unbroken drill cores gathered at depth support the observation of few flowing test sections in the deeper bedrock (Figure 8-51 in the **Site description Forsmark**).*

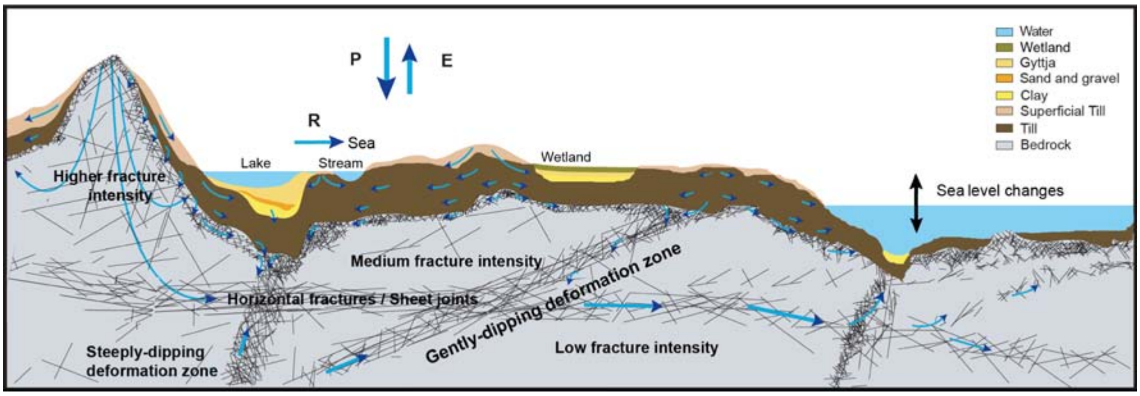


Figure 4-18. *Cross-section cartoon visualising the notion of a shallow bedrock aquifer and its envisaged impact on the groundwater flow system in the uppermost part of the bedrock within the target area. The shallow bedrock aquifer is probably hydraulically heterogeneous but at many places it is found be very anisotropic causing a short circuit of the recharge from above. The shallow bedrock aquifer is conceived to constitute an important discharge horizon for the groundwater flow in outcropping deformation zones. P = precipitation, E = evapotranspiration, R = runoff (Figure 3-21 in /Follin et al. 2007a/).*

Confidence

In general, there is a high confidence in the bedrock hydrogeological model of the site and the assignment of hydraulic properties. The main reason for this assertion is the consistency between different types of hydraulic data, which all support the presence of anisotropic hydrogeological conditions in the area. These conditions are: 1) high transmissivity in the gently dipping fracture zones outside the target volume, 2) few flowing fractures at depth in the target volume (FFM01 and FFM06), and 3) a highly transmissive system of fractures, including sheet joints, in the near-surface realm inside the target volume (FFM02), which are connected over long distances. The various types of hydraulic data are also consistent with the understanding of the geology and the rock stresses at Forsmark and are also supported by the groundwater chemical data (see Section 4.8).

Uncertainties remain in the hydraulic properties of deformation zones and fractures south-west and north-east of the tectonic lens and target volume. The most important of these concerns the hydraulic character of the regional Singö deformation zone close to the north-eastern border of the tectonic lens, and the hydraulic properties further downstream of the target volume. Although interference tests conducted show no hydraulic contact between the north-eastern part of the target volume and the bedrock on the north-eastern side of the Singö deformation zone, where the SFR repository is situated, it cannot be excluded that such hydraulic connections exist at other locations along this regionally significant zone.

The implementation of the geometry and hydraulic properties of fracture domains and deformation zones and the shallow bedrock aquifer into a numerical model have revealed that the model can be matched to results of large-scale interference tests, to observed natural point-water heads in percussion holes and soil pipes, and to the concentrations of various species measured in groundwater samples from boreholes. However, there is some concern whether the measured groundwater levels (point-water heads) are natural or disturbed, e.g. disturbed by the abstraction of drainage water at SFR. Even though the abstraction of drainage water in the SFR repository is an uncertain boundary condition that may affect the natural groundwater levels, it is concluded that the hydraulic stresses (drawdowns) induced by the cross-hole tests run in the target volume are sufficiently strong to allow for a fair calibration of the hydraulic properties. The magnitudes of the hydraulic gradients derived from the groundwater flow model using these calibrated properties are quite reasonable, whereas the gradients inferred from the field measurements of the natural groundwater flow in deep boreholes are much higher and generally exceed the maximum possible topographic gradient at the site by orders of magnitude. Conversely, *in situ* measurements of flow rates in fractures by tracer dilution techniques give flow rates that are sometimes much larger than what is expected considering the apparent transmissivity of the same fracture and reasonable hydraulic gradients. The reason for this behaviour is not fully understood, although is likely to be related to the simplifying assumptions necessary to make such scoping calculations on the level of individual flow conductors. The flow rates as a whole, however, are comparable to hydraulic modelling results obtained using the hydraulic network models.

4.7 Integrated fracture domain, hydrogeological DFN and rock stress models

Fracture domain FFM03 in the hanging wall to the gently dipping zones ZFMA2 and ZFMF1 (Figure 4-19) was defined on the basis of an inferred even distribution of open fractures with depth and is contiguous with a bedrock volume containing abundant gently dipping fracture zones. This contrasts sharply with the bedrock further to the north-west in the footwall to these gently dipping zones. Modelling work has shown that connected open fractures are also more evenly distributed in the hanging wall to the gently dipping zones A2 and F1 (Figure 4-19). Furthermore, the maximum horizontal stress is reduced in this domain relative to that in the bedrock volume further to the north-west (Figure 4-19). The even distribution of open fractures and the occurrence of gently dipping fracture zones are consistent with a general stress-released volume. Indeed, the fracture characteristics may have inhibited a more significant build-up of horizontal stresses in this bedrock volume.

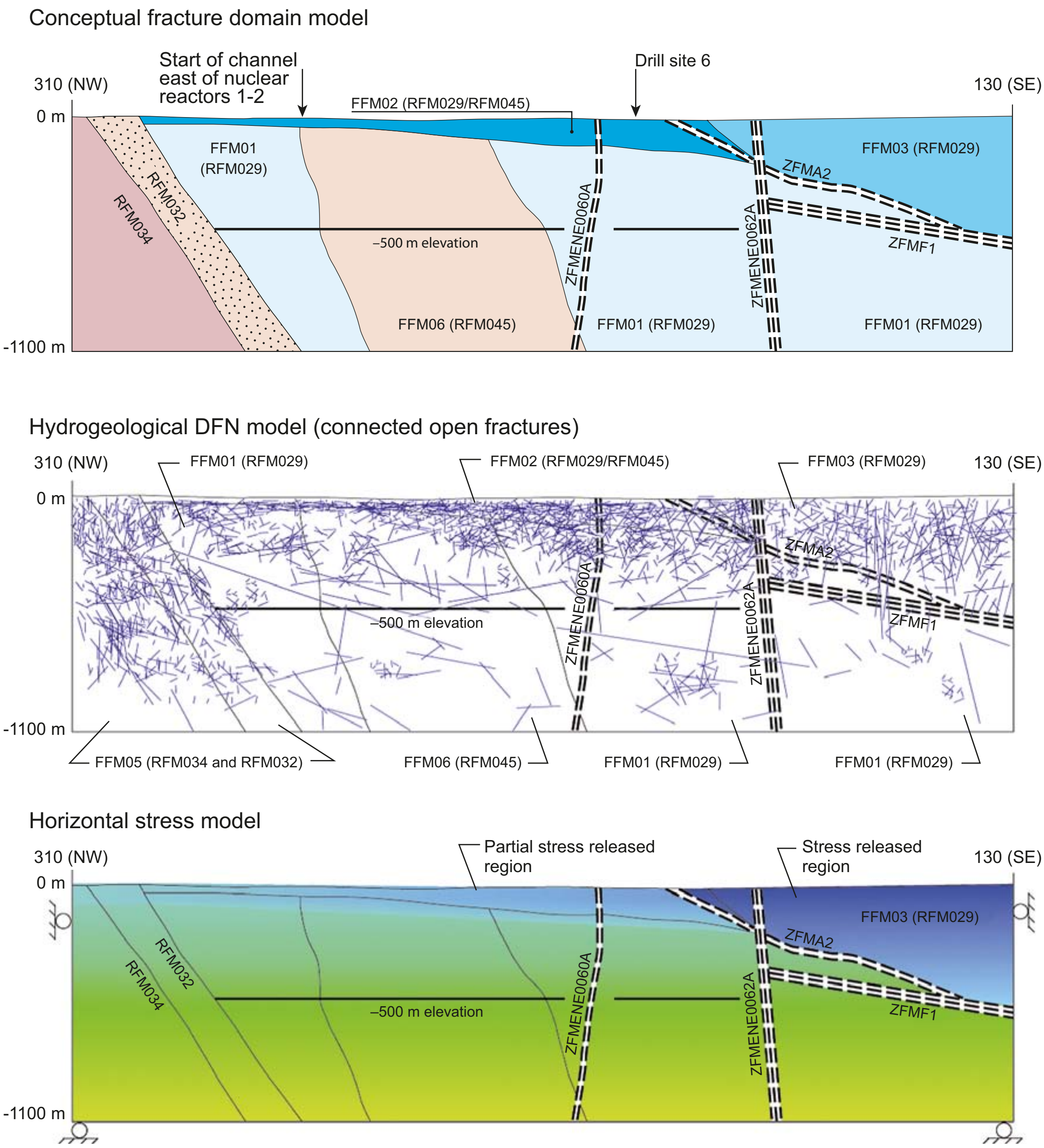


Figure 4-19. *Comparison of fracture domain, hydrogeological DFN and maximum horizontal stress models along a NW-SE profile in the north-western part of the candidate volume (Figure 11-21 in the* ***Site description Forsmark****).*

The bedrock in the footwall to the gently dipping zones ZFMA2 and ZFMF1 (Figure 4-19) is far more inhomogeneous especially with respect to the distribution of open fractures. This attribute, together with some consideration for increased alteration (albitization) in fracture domain FFM06, formed the basis for the recognition of three different fracture domains in this volume (FFM01, FFM02 and FFM06). An increased intensity of connected open fractures and reduced maximum horizontal stress characterise the upper part of the bedrock in the footwall to zone A2, and this volume corresponds more or less to fracture domain FFM02 (Figure 4-19). A striking consistency between the different models concerns how the intensity of connected open fractures and the stress-released volume conform to the increased thickness of domain FFM02 as the gently dipping zone A2 is approached (Figure 4-19). It is suggested that this change with respect to zone A2 is related to the increased frequency of sub-horizontal and gently dipping fractures in the vicinity of this zone. Since these structures are oriented at a high angle to the vertical stress, which corresponds to the minimum principal stress, there is a more favourable environment for the modification of the aperture of ancient fractures and even formation of new stress-release joints and, consequently, the more pronounced release of stress closer to the A2 zone.

In contrast, the major part of fracture domains FFM01 and FFM06 inside the target volume, including especially the repository volume around 500 m depth, is characterised by relatively few open fractures, a compartmentalised pattern close to the percolation threshold for the connected open fractures and higher maximum horizontal stress. It is inferred that the low intensity of open fractures favoured a more significant build-up of horizontal stresses. The character of the connected open fractures in this volume implies restricted groundwater circulation.

In summary, a bedrock volume where there is a limited number of open fractures and, by corollary, a low permeability is consistent with higher rock stresses. However, the properties of such bedrock are affected by the stress-released conditions in the near-surface realm and close to gently dipping ancient structures.

4.8 Groundwater

4.8.1 Evolution during the Quaternary period

Several water types which are now present in the bedrock can be associated with past climatic events during the Pleistocene (c. 2 Ma), including inter-glaciations, glaciations, deglaciations, and associated changes in the shore level in connection with transgressions and regressions. Among these, the last glaciation and the post-glacial period (the Holocene starting 10,000 years ago) are the most important for the groundwater development in the Fennoscandian Shield, especially in terms of land uplift and shoreline displacement, as well as the development of the Baltic Sea.

The post-glacial development during the Holocene (Figure 4-20) reveals that when the continental ice melted and retreated from the Forsmark area around 8800 BC, glacial meltwater was hydraulically injected under considerable pressure into the bedrock. The exact penetration depth is unknown and is also dependent on the geometry and heterogeneity of the system. However, chemical and isotopic data from porewater in the rock matrix indicate penetration depths of about 550 metres southeast of the target area adjacent to the gently dipping deformation zone A2, whereas no obvious cold climate signature has been found at these depths in the low-conductive target volume of the rock. Since the deglaciation of the Forsmark region coincided with the end of the Yoldia period, there are no signs of Yoldia Sea water in the bedrock. The Ancylus Lake (8800 to 7500 BC) was lacustrine and developed after the deglaciation. This period was followed by the brackish Littorina Sea (after 7500 BC). During the Littorina Sea stage, the salinity was considerably higher than at present, reaching a maximum of about 15‰ in the period 4500 to 3000 BC. Dense brackish seawater from the Littorina Sea penetrated the bedrock, resulting in a density intrusion that affected the groundwater in the more conductive parts of the bedrock. When the first parts of the Forsmark region subsequently emerged from the sea, starting c. 500 years BC, recharge of meteoric water subsequently formed a freshwater layer on top of the saline water because of its lower density. As a result of the flat topography of the Forsmark area and of the short period that has elapsed since it emerged from the sea, the out-flushing of saline water has been limited, and consequently a freshwater layer is only present at shallow depth.

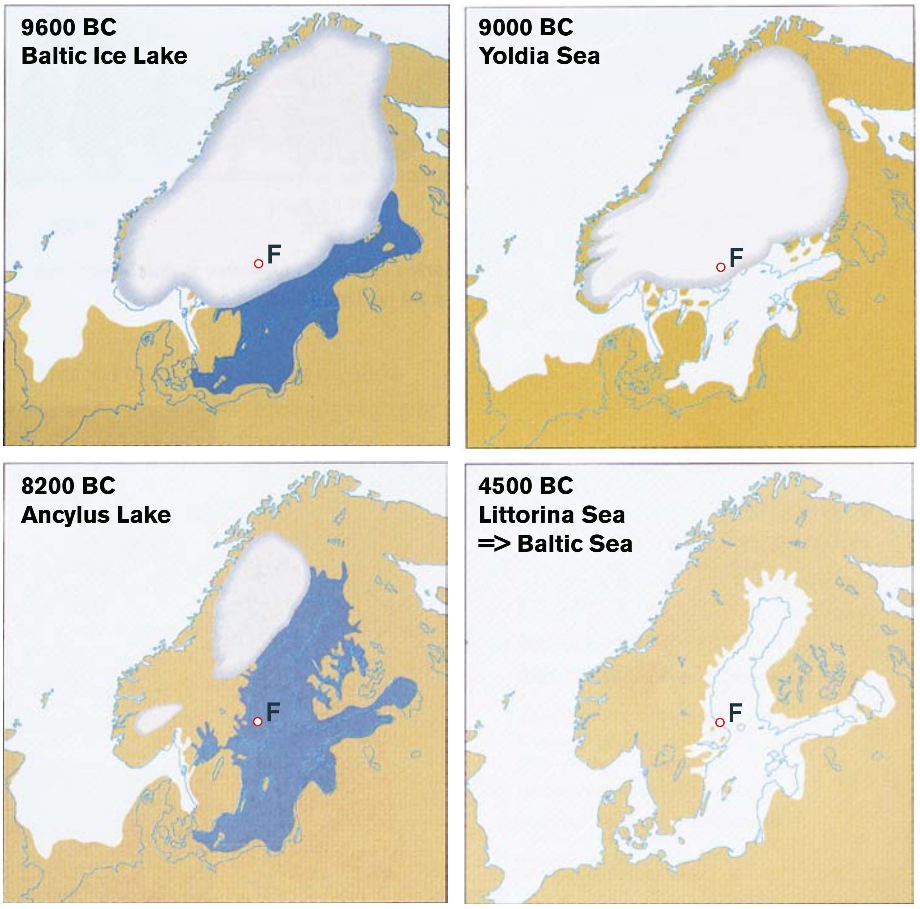


Figure 4-20. *Map of Fennoscandia with some important stages during the Holocene period. Four main stages characterise the development of the aquatic systems in the Baltic basin since the latest deglaciation: the Baltic Ice Lake (13,000–9500 BC), the Yoldia Sea (9500–8800 BC), the Ancylus Lake (8800–7500 BC) and the Littorina Sea 7500 BC–present). Fresh water is symbolised with dark blue and marine/brackish water with pale blue. The Forsmark area (notated 'F') was probably at or close to the rim of the retreating ice sheet during the Yoldia Sea stage (Figure 11-22 in the* ***Site description Forsmark****).*

The Quaternary evolution has affected the groundwater chemistry at Forsmark, especially in the more conductive parts of the bedrock, but changes in chemistry are not restricted to post-glacial time. There is groundwater and porewater evidence that indicates the presence of old (pre-Holocene) meteoric water that originated during a warm climate. The age of this component is unknown, but possibilities include pre-Pleistocene or even older. The hydrogeochemistry of the Forsmark area cannot be explained without recognising this older component. The present groundwaters therefore are a result of mixing and reactions over a long period of geological time. The interfaces between different water types are not sharp due to molecular diffusion, but reflect variability in the structural-hydraulic properties.

4.8.2 Groundwater composition and water – rock interactions

Explorative analyses of groundwater chemistry data measured in samples from cored boreholes, percussion boreholes and soil boreholes, and hydrogeochemical modelling have been used to evaluate the hydrogeochemical conditions at the site in terms of origin of the groundwater and the processes that control the water composition. Although the data set is rather limited, the results have revealed that the current groundwater composition in general supports the occurrence of different hydrogeological regimes in the candidate volume, i.e. the bedrock at depth in the target volume, the uppermost part of the bedrock in the target volume and the rock south-east of the gently dipping zone A2 with its swarm of gently dipping fracture zones.

Groundwater in fractures

Groundwaters in the uppermost 100 to 200 m of the bedrock display a wide range of chemical variability, with chloride concentrations in the range 200 to 5,000 mg/L (Figures 4-21 and 4-22) suggesting influence of both brackish marine water (i.e. recent Baltic or old Littorina Sea relicts) and meteoric waters. In the bedrock in the footwall to zone A2 including fracture domain FFM02, this shallow system is controlled by flow along highly transmissive, sub-horizontal fractures and sheet joints (i.e. the shallow bedrock aquifer), which is still in the process of flushing out residual brackish marine (Littorina) groundwaters. Furthermore, a sharp decrease in tritium content at about 150 m depth, as well as carbon-14 data (Figure 4-21), indicate that these shallow groundwaters have short residence times that are in the order of only a few decades to a few hundred years.

At depths greater than c. 200 m, the water composition is indicative of brackish marine water with chloride concentrations in the range 2,000 to 6,000 mg/L and with a clear Littorina Sea component (Figures 4-21 and 4-22), as indicated by concentrations of magnesium and the ratio of bromide to chloride concentrations. This water type is recognised down to 600 to 700 m depth in the transmissive gently dipping fracture zones in the south-eastern part of the candidate volume, contiguous with fracture domain FFM03, whereas the penetration depth in fracture domain FFM01 in the target volume, where the frequency of water-conducting fractures is low, is restricted to c. 300 m. Below these depths in FFM01, the water composition indicates brackish to saline non-marine groundwaters (i.e. an absence of Littorina Sea influence), reflecting processes which have occurred prior to the intrusion of the Littorina Sea waters. These deep waters further show an increase in calcium with depth, which is a well recognised trend and indicative of water/rock interactions that occur under increasingly low flow to stagnant groundwater conditions with increasing depth.

Composition of rock matrix porewater

Analyses of the composition of rock matrix porewater also support the occurrence of low groundwater turnover in fracture domain FFM01. Porewater from this domain generally has a lower chloride content and is enriched in oxygen-18 compared with the fracture groundwaters, indicating a transient state between the porewater and groundwater down to at least 650 m depth (Figure 4-22). A signature with low chloride, low magnesium and enriched in oxygen-18 has been preserved far away from conducting fractures, suggesting that these porewaters have evolved from an earlier, very long lasting circulation of old dilute groundwaters in a few fractures. This is also consistent with the still prevailing transient state between this porewater and fracture groundwaters from equivalent depths which, based on chlorine-36 and helium-4 dating, have residence times of more than 1 Ma.

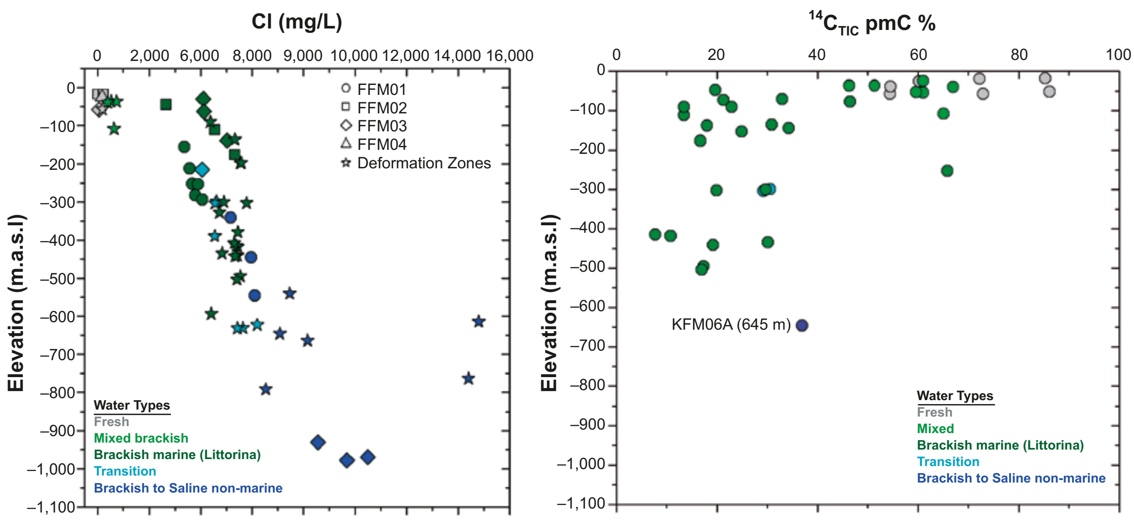


Figure 4-21. *Left: Cl⁻ concentration as a function of elevation as measured in groundwater from the different fracture domains and from deformation zones (from Figure 9-5 in the **Site description Forsmark**). Right: $^{14}C_{TIC}$ as a function of elevation. The only sample of brackish non-marine groundwater type analysed indicated contamination during sampling (Figure 9-20 in the **Site description Forsmark**).*

South-east of the target volume in the hanging wall rock to zone A2, a situation close to steady-state is suggested between porewater and fracture groundwater down to about 200 m below the surface, reflecting the high frequency of water conducting, gently dipping fracture zones, and the rapid circulation of significant volumes of water in this area (Figure 4-22). At greater depth, the porewater has lower chloride contents than the fracture groundwater indicating a transient state down to approximately 650 m depth. The lower chloride content and an isotope signature increasingly depleted in ^{18}O with increasing depth indicate that the porewater in these gently dipping zones probably stores a dilute water with a cold-climate signature. The significantly negative $\delta^{18}O$ values preserved far from the water-conducting fractures indicates that this cold-climate signature relates to glacial water circulating for a considerable period in the fractures at these depths. Since the last deglaciation, the porewater signature has become overprinted with a Littorina- and/or a Baltic-type signature as indicated by chloride, magnesium and oxygen-18 in porewaters sampled closer to the conducting fractures.

On-going reactions

Whilst mixing (both advection-dispersion and molecular diffusion) is the major process giving rise to present-day groundwater compositions, the role of reactions has also been addressed. In particular, the alkalinity and redox buffering capacity of the bedrock is of key importance for groundwater composition and future changes due to, for example, potential infiltration of dilute and oxidised water.

The presence of limestone (calcite) and extensive biogenic activity in the Quaternary overburden give rise to pH values usually above 7, calcium concentrations mostly between 50 and 200 mg/L and bicarbonate concentrations in the range 200 to 900 mg/L in the near-surface waters (down to c. 20 m depth). Concentrations then decrease to very low values at greater depths. However, bicarbonate is

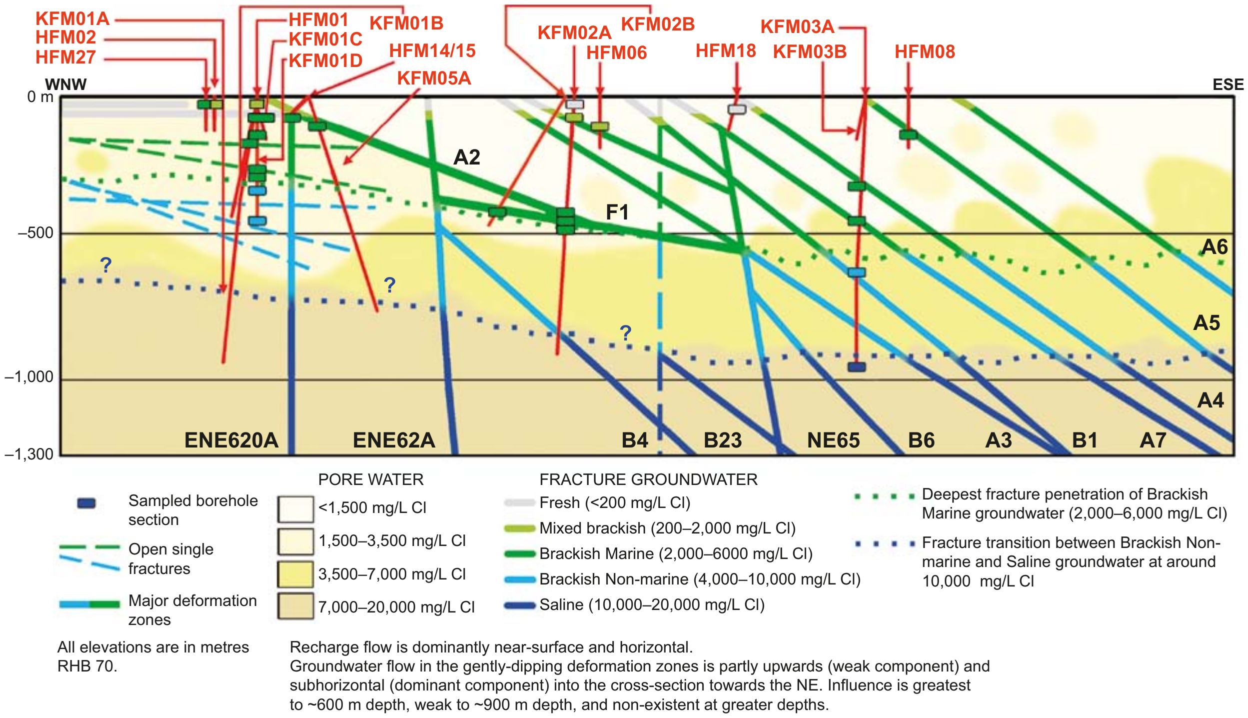


***Figure 4-22.** Illustration of the groundwater composition along a WNW-ESE cross-section through the candidate area at Forsmark as interpreted from hydrogeochemical data. The location of the boreholes and the sections that have been sampled are shown together with the main fracture groundwater types that characterise the site. The chloride distribution with depth along the fracture zones and single open fractures, as well as the subdivisions of the rock matrix porewater based on chloride concentration are also shown. The dotted lines in different colours crossing the section represent the approximate depths of penetration of the major groundwater types along hydraulically-active fracture zones (Figure 11-23 in the **Site description Forsmark**).*

relatively high in most of the brackish marine groundwaters hosted in the upper 600 m of the gently dipping fracture zones south-east of the target volume, whereas brackish non-marine groundwaters below 300 m in fracture domain FFM01 have low bicarbonate contents.

The pH buffering capacity in Forsmark groundwaters at depths greater than 100 m appears to be controlled by the calcite system, and modelling indicates that this water is in equilibrium with calcite. Investigation of fracture minerals shows that calcite in fractures is abundant and that no extensive leaching has occurred in response to past glaciation/deglaciation events.

According to data analyses and modelling of the redox system, reducing conditions currently prevail at depths greater than c. 20 m. Most of the Eh values determined in brackish groundwaters (at depths between 110 and 646 m) seem to be controlled by the occurrence of an amorphous iron oxyhydroxide with higher solubility than a truly crystalline phase. This indicates that the iron system is disturbed. This conclusion is supported by mineralogical investigations that have identified the presence of fine-grained amorphous to poorly crystalline phases now evolving towards more crystalline phases. Dissolved sulphide concentrations are systematically low, possibly due to the precipitation of amorphous Fe(II)-monosulphides, linked to the activity of sulphate-reducing bacteria (SRB). At depths greater than 600 m, the dissolved sulphide concentrations increase, which is consistent with the occurrence of SRB and with the active precipitation of Fe(II)-monosulphides. The iron system at these depths seems to be limited by crystalline oxides, mainly hematite.

Elevated concentrations of uranium have been detected in groundwaters associated with a Littorina Sea component and the highest concentrations are found in waters in the gently dipping fracture zones south-east of the target volume. There are indications that these elevated concentrations are related to easily dissolvable uranium fractions in fracture coatings in contact with these waters. Speciation-solubility calculations support this conclusion and indicate that the high uranium contents are the result of the control exerted by an amorphous (and very soluble) uranium phase present in the system, and the weakly reducing Eh values that may allow uranium complexation and re-equilibration depending on Eh and dissolved carbonate.

The presence of goethite (FeOOH) in some hydraulically active fractures and fracture zones in the upper part of the bedrock, mainly within the gently dipping fracture zones A2 and F1, indicates circulation of oxygenated fluids during some period in the past (potentially during the Quaternary). However, the presence of pyrite in the same zones suggests that the circulation of oxygenated fluids has been concentrated along channels in which different redox micro environments may have been formed. Mobilisation as well as deposition of uranium in the upper 150 m of the bedrock is indicated by U-series decay analyses of fracture coatings.

The analyses of the current redox system at Forsmark have consistently indicated that sampling (or drilling-induced) perturbation may have altered the original redox conditions of the hydrogeochemical system. Examples include oxygen intrusion and precipitation of amorphous iron oxyhydroxides, as indicated by the colloidal composition (see section below) and mineralogical determinations. Additionally, there could have been modification of the original Eh and/or alkalinity by drilling waters, and an increase in dissolved uranium contents and changes in sulphide contents could have been caused by one or more of these disturbances. Despite these potential disturbances, the buffer capacity of the system maintains a substantially reducing character. As far as the potential redox buffering capacity of the fracture system is concerned, it is concluded that previous oxidising episodes have not been intense enough to exhaust the reducing capacity of fracture-filling minerals, which are still present in the shallow system (for example chlorite and pyrite). Any potential build up of reducing capacity in the fracture minerals during recent periods of reducing groundwater conditions is difficult to estimate, but it can be concluded that the amounts of recent (Quaternary) minerals formed is very small.

Dissolved gas and colloids

Analyses of gas dissolved in groundwater at Forsmark have shown that the gas content increases with depth, but the waters are far from being oversaturated by gas at the depths from which they were sampled. The major gas components are nitrogen and helium. Methane has also been detected, but generally in small amounts (less than 0.2 mL/L). Currently, it is not known whether the methane is of biogenic or non-biogenic origin.

Colloid amounts in Forsmark groundwater are comparable to those found in other granitic environments. The colloids are composed mainly of iron and sulphur compounds. Uranium associated with the colloids has been found in boreholes KFM02A and KFM06A, in line with the high groundwater uranium concentrations found in these boreholes. The uranium content of the colloids is approximately 10% of the uranium concentration in the groundwater and colloidal transport is, therefore, a result, but not the origin, of the high uranium content in the groundwater.

Confidence

There is generally high confidence in the description and understanding of the current spatial distribution of groundwater composition, mainly due to the consistency between different analyses and modelling of the chemical data, but also due to the agreement with the hydrogeological and structural geological understanding of the area. Furthermore, the existence of a near-surface redox reaction zone appears to be well-established, even if there is uncertainty in the data interpretation, and the common occurrence of calcite suggests that there is buffering capacity against the effects of penetration of dilute groundwater. One important remaining uncertainty concerns the increase in sulphide concentrations measured in the on-going monitoring programme. Initial drilling and pumping may have disturbed the system or may have facilitated sulphate reduction. Details of the groundwater composition and associated uncertainties and the implications for SR-Site are provided in the **Data report**, Section 6.1.

4.8.3 Groundwater flow and consistency with groundwater signatures

The palaeo-evolution of groundwater composition at Forsmark during the last c. 10,000 years, i.e. during the Holocene, has been simulated and the results compared with measured concentrations of different elements in the boreholes. The conceptual model for the present-day distribution is illustrated in Figure 4-23 and should be compared with the hydrogeochemical site descriptive model described in Figure 4-22.

The initial conditions for the palaeo-hydrogeological simulations that start at 8000 BC are defined in terms of the presence of mixtures of different reference waters in the bedrock. These conditions are derived on the basis of analyses of present-day water composition in fractures and matrix porewater, which reveal that there must have been old meteoric waters derived from both warm and cold climate events in the bedrock in the Forsmark area before the injection of *Glacial Melt Water* during the last deglaciation just prior to the Holocene.

The results of the simulation with the hydrogeological model show fair agreement with measured concentrations of chloride, the bromide/chloride ratio, $\delta^{18}O$ and bicarbonate concentrations in the boreholes (Figure 8-46 to 8-50 in the **Site description Forsmark**). Furthermore, the model predicts deeper penetration of Littorina Sea water along the gently dipping fracture zones intersecting fracture domain FFM03 south-east of the target volume (down to c. 500 to 600 m depth) than in steeply dipping deformation zones intersecting fracture domain FFM01 inside the target volume (down to c. 300 to 400 m depth). Although not matching the exact concentrations, the model also predicts higher salinity in the fracture water than in the rock matrix porewater, which is consistent with the field observations (Figure 4-22).

The palaeo-hydrogeological simulations also support the hydrochemical observations in surface water and shallow groundwater, which indicate that there is probably no ongoing discharge of deep saline water into the freshwater surface system within the area covered by the well-connected and highly transmissive network of fractures and sheet joints in the uppermost c. 150 m of the bedrock. Furthermore, the simulation results are consistent with the field observations of relict marine remnants, which also include deep saline signatures, in the groundwater at relatively shallow depths in the Quaternary deposits in restricted areas outside this network of structures in the upper part of the bedrock. One such area is Lake Gällsboträsket, which coincides with the Eckarfjärden deformation zone (see also Section 4.10).

A comprehensive uncertainty analysis with focus on hydraulic parameter heterogeneity within the target volume was performed and the results demonstrate that model calibration against hydrochemical data is sensitive to parameter heterogeneity in the bedrock hydrogeological properties, which is expected in a sparsely fractured rock mass.

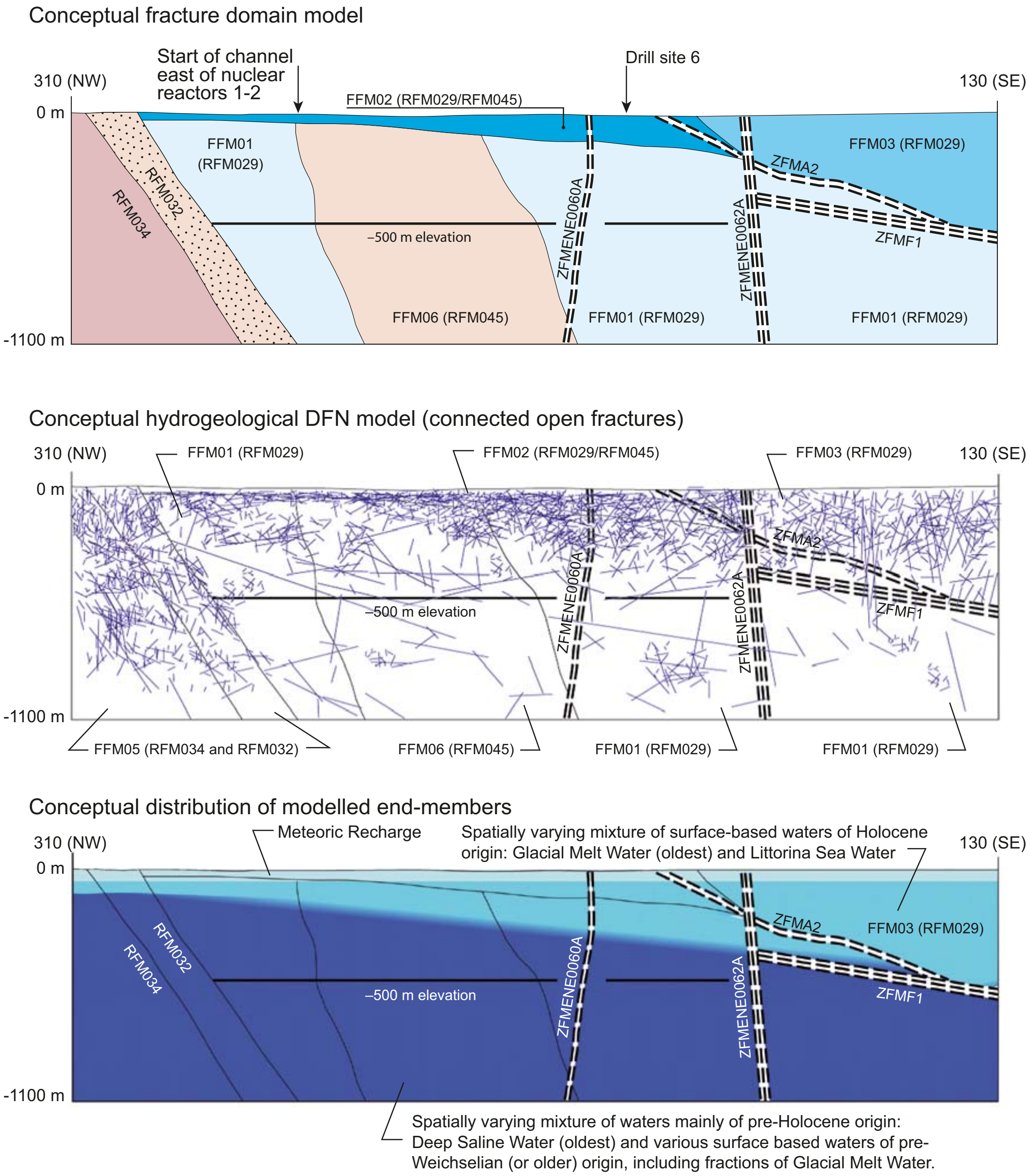


***Figure 4-23.** Comparison of conceptual models for fracture domains, hydrogeological DFN and the present-day distribution of different end members along a NW-SE profile in the north-western part of the candidate volume (Figure 11-25 in the **Site description Forsmark**).*

4.9 Bedrock transport properties

4.9.1 Rock matrix properties

The bedrock retardation model in the site description is comprised of a qualitative identification and description of typical fracture types and deformation zones at the Forsmark site with regard to processes of relevance for transport of environmental solutes and radionuclides. Furthermore, it builds on quantitative data describing material properties and the relative abundance and spatial distribution of the different geological materials comprising the rock matrix, as well as relevant alteration types and secondary minerals found in association with fractures and deformation zones.

The input data for the retardation model consist of formation factors for solute transport in the rock matrix, matrix porosities, specific surface areas of internal micro-surfaces, cation exchange capacities

(CEC), and sorption properties of rock in contact with synthetic groundwater of varying composition. Formation factors are obtained from *in situ* measurements under natural stress conditions, derived from high spatial resolution geophysical logging in the site investigation boreholes, together with data from laboratory studies.

The limited data available for the different rock masses indicate that there are generally no significant differences in the retardation properties between the rock types present in the rock domains RFM029 and RFM045. Furthermore, taking variations in the parameter values into account, there are very few indications of significant differences between the different rock types. As far as key fracture classes are concerned, either a thin layer of fracture coating or no coating at all have been identified as being of importance, since these classes are likely to be associated with a lower degree of retardation.

The deformation zone structural elements for which retardation properties have been tabulated are all comparatively heterogeneous in their structure. From the material properties data, these structural elements are identified as potentially strong sinks for radionuclides transported from a leaking repository. However, this should be considered in the context of the possibly lower hydrodynamic transport resistance of these zones set against their increased micro-structural complexity including additional diffusion-accessible surface area.

Confidence

The porosity and effective diffusivity of the rock matrix appear to be relatively well constrained for the main rock types. Although there is evidence for *in situ* compression of pore spaces in the unaltered matrix rock, the effect upon material properties can be quantitatively bounded. Sorption uncertainty is only semi-quantitatively established and is large for many species and specific groundwater compositions. Sorptivities of certain radionuclides in contact with specific groundwaters and rock types are not known or are not supported by sufficiently large sample sizes to be considered statistically quantified. Sorptivities of U and Np under strongly reducing conditions are likely to be underestimated owing to difficulties in maintaining appropriately low redox conditions in the laboratory. There is uncertainty in the distribution and thickness of altered rock surrounding flow paths. However, the importance of this feature depends on the differences in the material properties of these materials relative to unaltered rock. The data suggest that altered rock is generally associated with increased retention for most radionuclides and therefore this uncertainty can be bounded by the retention properties of unaltered rock. A further quantification of the retardation properties and their associated uncertainties for the purpose of the SR-Site assessment is provided in the **Data report**, Section 6.7.

4.9.2 Flow related transport properties

Flow related transport properties are evaluated as part of the SR-Site assessment, see Section 10.3.6. However, as a support to the selection of processes and parameters to be considered in the safety assessment modelling, analyses of flow related transport properties were also carried out as part of the site-descriptive modelling and reported in the **Site description Forsmark**.

Molecular diffusion and sorption within the rock matrix is considered to be an important transport retardation mechanism limiting the migration of escaping radionuclides. A key parameter describing this process is the ratio of the surface area in contact with flowing water (the flow-wetted surface) and the flow rate. In general, the greater surface area in contact with flowing water for a given flow rate, the greater the interaction will be with both the fracture surface itself and the rock matrix. The flow-wetted surface to flow ratio is commonly referred to as the flow related transport resistance or (F), see Section 10.3.6 for details.

Numerical simulations of flow and migration provided indications of median F values for "typical" flow paths on the 100 m scale to be on the order of 10^6 y/m at depths larger than 400 metres in fracture domain FFM01. The simulations further demonstrate that less than 4% of the hydrogeological DFN realisations for FFM01 below 400 m depth exhibit hydraulic connectivity of any kind. Simulations of other fracture domains (upper regions of FFM01, FFM02, and FFM03) give indications of substantially smaller F values as compared with FFM01 at depths below 400 m. An analytical model analysis reveals that flow paths within gently dipping zones such as ZFMA2 are associated with F values of less than 10^3 y/m for transport from repository depth to the near surface. Steeply dipping deformation zones, on the other hand, (for example zone ZFMENE0060A) are found to have F values on the order of 10^5 y/m.

Scoping calculations of the effects of flow channelling on the interpretation of hydraulic borehole data indicate that channelling would not be severe enough to cast doubt on the utility of the hydrogeological models produced based on the site data. It is further concluded that additional physical mechanisms enhancing solute uptake, such as radial diffusion from channels of limited extent and diffusion into stagnant zones with concomitant matrix diffusion, may increase transport retardation substantially. Flow channelling may therefore possibly have an overall beneficial effect.

Confidence

Uncertainty in hydrogeological DFN parameters and the role of channelling phenomena may lead to underestimation of flow channel frequency in the target volume. However, the overall F values for typical flow paths through the repository volume should not be greatly different, provided that the fracture transmissivity model is determined to be reasonable (i.e. it yields the approximately correct order of magnitude in flow predictions). The hydrogeological DFN fitting parameters for fractures within the repository volume can only be properly constrained by statistics from mapping of open fractures in tunnels. Nevertheless, it is judged that the current hydrogeological DFN, with its alternative descriptions of the transmissivity size correlation, provides adequate bounds on the uncertainty of the flow-related transport properties. These uncertainties are further discussed in the **Data report**, Section 6.6.

4.10 The surface system

The Forsmark area is special in many ways and does not represent a typical coastal Swedish site located at the shoreline of the Baltic Sea. Post-glacial land uplift, in combination with the flat topography, implies fast shoreline displacement, which has resulted in a very young terrestrial system that contains a number of newborn lakes and wetlands. Shallow and with sediments rich in calcium, the oligotrophic hardwater lakes are unique for northern Uppland.

4.10.1 Evolution during the Quaternary period

The character of the current surface system has been strongly influenced by the climate evolution during the Quaternary period and the Forsmark area has repeatedly been covered by glacier ice. In total there have been around 50 glacial/interglacial cycles during the Quaternary and there is evidence for at least two periods during the Weichsel when large parts of Sweden were free of ice. Deposition of sediments at Forsmark can be linked to the Weichselian glacial phase and during and after the following deglaciation. In the western part of the candidate area, a silty-clayey till with an extremely high degree of consolidation is present. Based on its stratigraphical position and pollen composition, it is concluded that the deposit is older than the latest ice advance. In addition, sediment-filled open fractures beneath till in the upper bedrock have been recognised (Figure 4-24). The deposition of these sediments is dated back to a late stage of the glacial phase, when large amounts of sediment-loaded meltwater were concentrated below and within the retreating ice. These fractures formed or were reactivated during a late stage of the local deglaciation and are inferred to be sheet joints formed in connection with the release of stress in the bedrock. During further retreat of the ice, glacial clay was deposited in low topographic areas. The deposition of this glacial clay at Forsmark is dated to the Yoldia Sea stage.

The development of the Baltic Sea after the latest glaciation has been characterised by ongoing shoreline displacement. The highest shore level in north-eastern Uppland developed during the Yoldia Sea stage (9500–8800 BC) of the Baltic Sea development and was located c. 100 km to the west of Forsmark. At that time, the Forsmark area was covered by c. 150 m of water. Since then, the shoreline displacement has been continuously regressive and most of the Forsmark area has emerged from the Baltic Sea during the last 2,000 years. The shoreline displacement has had a large impact on the distribution and relocation of fine-grained Quaternary deposits. Wave washing and bottom currents have eroded, transported and re-deposited sand, gravel and post-glacial clay.

Figure 4-24. a) Laminated silt in an open fracture in the north-western part of the excavated area at drill site 5. The site was originally covered with till that has been removed. b) Horizontal fractures along the more than 1 km long canal between the Baltic Sea and the nuclear power reactors in Forsmark. These fractures are inferred to be sheet joints formed in connection with the release of stress in the bedrock (Figure 11-2 in the ***Site description Forsmark****).*

The recent emergence of the Forsmark area from the Baltic Sea implies that peat formation has affected the area for a relatively short period of time. The shoreline displacement continuously transforms sea bottoms to new terrestrial areas or to freshwater lakes. Lakes and wetlands are successively covered by fen peat, which at some locations is covered by bog peat. At Forsmark, rich fens form the dominant type of peat. Bogs do occur, but they are few and still young. Peat is found most frequently in the most elevated south-western part of the area, i.e. in the area that has been above sea level for a sufficiently long time for infilling of basins and for peat to form.

The post-glacial development of the ecosystems at Forsmark is strongly correlated to climate changes and the shoreline displacement, but also to human activities during the last c. 2,000 years, as discussed in Section 4.10.3.

Detailed investigations to evaluate the occurrence of palaeoseismic activity during the latest part of and after the Weichselian glaciation in and around the Forsmark area have been carried out in the context of the site investigation work. None of the morphological lineaments that have been recognised have been inferred to represent late- or post-glacial faults. Furthermore, no deformational features in Quaternary sediment have been unambiguously related to seismic activity. On the basis of these results, there is no evidence in the geological record for major (magnitude > 7 on the Richter scale) earthquakes.

4.10.2 Description of the surface system

The landscape in Forsmark is a relatively flat peneplain that dates back to the Precambrian time, i.e. prior to c. 540 million years ago. This peneplain dips gently towards the east. The candidate area (Figure 4-3) is almost entirely located less than 20 m above current sea level. Despite the modest topography, the upper surface of the bedrock is found to undulate over small distances implying large variations in the thickness of the Quaternary cover.

Quaternary deposits

More than 90% of the regional model area is covered by Quaternary deposits, with till as the dominant deposit, especially in the terrestrial part. Post-glacial clay, including clay gyttja, is predominantly found in the deeper parts of valleys on the sea floor and only minor occurrences have been documented in the terrestrial area. Post-glacial gravel and sand are frequently superimposed on glacial clay. The thickness of the Quaternary deposits is generally larger in the marine area (average c. 8 m) than in the terrestrial part (average c. 4 m). Clay gyttja is frequent in the surface of the wetlands located at low altitudes, e.g. along the shores of Lake Fiskarfjärden and Lake Gällsboträsket. Gyttja is formed in lakes and consists mainly of remnants from plants that have grown in the lake.

Lakes and water courses

The lakes at Forsmark are small (at most c. 0.6 km^2) and shallow, with maximum depths in the range 0.4 to 2 m. The largest lakes in the area are Lake Fiskarfjärden, Lake Bolundsfjärden and Lake Eckarfjärden (Figure 4-3). Flows of sea water into the most low-lying lakes have been registered during events of very high seawater levels. Furthermore, interpretation of data reveals that the lakes sometimes act as recharge sources to till aquifers in the riparian zone during summer, because of water losses from this zone by evapotranspiration. The annual precipitation and runoff are 560 and 150 mm, respectively.

No major water courses flow through the central part of the candidate area. The brooks downstream of Lake Gunnarsboträsket, Lake Eckarfjärden and Lake Gällsboträsket carry water most of the year, but can be dry for long periods during dry years such as 2003 and 2006. Many brooks in the area have been deepened for considerable distances for drainage purposes.

Hydraulic properties of Quaternary deposits

Hydraulic data show that the horizontal hydraulic conductivity within the till is significantly higher than the vertical conductivity and that the groundwater levels in the Quaternary deposits are shallow and closely correlated to the topography. These groundwater levels are significantly higher than the levels in the uppermost part of the bedrock within the target area, where the gradients in groundwater level are very small. This suggests that local, small-scale recharge and discharge areas, involving groundwater flow systems restricted to the Quaternary deposits, overlie the larger-scale flow systems associated with groundwater flow in the bedrock. In contrast, outside the target area and the tectonic lens, for example in the area around Lake Eckarfjärden, groundwater levels in the bedrock are well above those in the Quaternary deposits and imply that flow systems involving the bedrock may have local discharge areas.

The lake sediments and the underlying till have low vertical hydraulic conductivities. This is indicated by lake-water/groundwater level relationships and the presence of relict marine chemical signatures beneath the lakes, which contrast with fresh groundwaters in the riparian zone.

Hydrogeochemistry

The till and the glacial clay are rich in calcium carbonate ($CaCO_3$), originating from Palaeozoic limestone that outcrops on the sea floor north of the Forsmark area. This, together with the recent emergence of the area above sea level, affects the chemistry of both surface water and shallow groundwater, causing high pH and high contents of major constituents, especially calcium and bicarbonate. Furthermore, the surface waters are high in nitrogen and low in phosphorous. This is a characteristic feature of the oligotrophic hardwater lakes that are typical of the Forsmark area.

Interpretation of hydrochemical data from surface water and groundwater in Quaternary deposits supports the hydrological evaluation that discharge of deeper groundwater occurs around Lake Eckarfjärden. In addition, this interpretation suggests that water sampled at the edge of Lake Gällsboträsket also has a signature indicating an influence from deep saline water. Furthermore, a mass balance calculation of chloride suggests that there must be an additional source of chloride in the water to that stored in the Quaternary deposits, possibly from discharging deep groundwater. The discharge of more saline deep groundwater at Lake Gällsboträsket is consistent with results of the hydrogeological model that predicts discharge of deep groundwaters both at Lake Eckarfjärden and Lake Gällsboträsket (Section 4.8). Both these lakes lie along the outcrop of the regional Eckarfjärden deformation zone (see Section 4.4).

Terrestrial ecosystems

The location of the Forsmark regional model area close to the sea makes the seashore a prominent feature in the east along with conifer forests, shallow lakes, mires and some agricultural land (Figure 4-4). The terrestrial vegetation is strongly influenced by the characteristics of the Quaternary deposits and by human land use. The calcareous influence is manifested in the flora by herbs and broad-leaved grasses, along with a number of orchid species. The long history of forestry in the area is seen today as a fairly high percentage of younger and older clear-cuts in the landscape.

Forests cover 73% of the land area at Forsmark and are dominated by Scots pine and Norway spruce growing mainly on wave-washed till. Wetlands are frequent and cover 10 to 20% of the three major delineated catchment areas. A major part of the wetlands comprises coniferous forest swamps and open mires. Agricultural land covers hardly 5% of the land area and is mainly located in the south-eastern part of the candidate area. It consists of arable land and grasslands. Some arable land and to a large extent semi-natural grasslands (intensively used grassland with a long management tradition) in the area have been abandoned following the nation-wide general regression of agricultural activities during the past 60 years.

The quantification of pools and fluxes of carbon and other elements has revealed that the vegetation is the largest store for organic material. The soil also accumulates organic material, but in much smaller quantities. The exception is the wetlands which are of significant importance for accumulation of organic matter and other elements, such as phosphorus, in the soil organic pool. In particular, the reed-dominated wetlands surrounding many of the lakes accumulate large amounts of organic matter and accompanying elements. This wetland type is one step in the succession of a lake to a terrestrial area. The export of carbon and organic matter is low in comparison with the internal fluxes of terrestrial areas.

Lake ecosystems

The lakes are all shallow and classified as oligotrophic hardwater lakes, i.e. they contain high calcium levels, but low levels of nutrients, as phosphorus is precipitated together with the calcium. These characteristics have a strong impact on the limnic ecosystem. Due to the shallow depth, all lake bottoms are reached by sunlight, and vegetation occurs at all depths. The dominant vegetation is stoneworts, which harbour various kinds of benthic fauna and also function as refuges for smaller fish. Common fish species are perch and roach, as well as tench and crucian carp. This last species survives low oxygen levels and is the only fish species present in the smaller lakes, where oxygen levels can be very low during winter.

Modelling results show that, contrary to typical Swedish lakes, primary production exceeds respiration in many lakes in the Forsmark area. Primary production in the larger lakes involves large amounts of carbon compared with the amounts entering the lakes from the surrounding catchment. Consequently, there is a large potential for carbon entering these lakes from the surroundings to be incorporated into the lake food web. However, according to the modelling, only a minor portion (7–10%) of the carbon incorporated into primary producers is transported upwards in the food chain. This means that most of the carbon incorporated into primary producers circulates within the microbial food web and is transported back into the abiotic carbon pools. In the larger lakes, there is a large degree of sediment accumulation and this sediment can be a permanent sink of pollutants and radionuclides.

Marine ecosystems

The marine ecosystem in the Forsmark area is relatively productive in a region of otherwise fairly low primary production. This is due to up-welling along the mainland. The salinity of the seawater is low (c. 5‰) due to large freshwater supply. The low salinity strongly affects the marine environment, as few organisms are adapted to this brackish condition, but rather to either freshwater or saltwater. Therefore, a mix of few freshwater and marine species is found in the Forsmark area. The marine biota in the area is dominated by benthic organisms such as macroalgae, vascular plants and benthic microalgae. Detrivores, snails and mussels feeding on dead material, dominate both hard and soft bottom substrates. The fish community is dominated by the marine species herring in the pelagic area, whereas limnic species, especially eurasian perch, dominate in the coastal areas and in the secluded bays.

Modelling results show that transport from land, lakes and streams gives only a minor contribution of organic matter to the marine ecosystem. The major fluxes of organic matter in the marine ecosystem are governed by water movements, and the advective flow of carbon is several orders of magnitude larger than any other flux, such as photosynthesis by primary producers, runoff from the adjacent terrestrial environment and burial. Even though parts of the coastal area are heterotrophic, the mean character of the whole area is autotrophic, i.e. more carbon is fixed in biomass by primary producers than is mineralised by all organisms. The major pool of carbon in the ecosystem is the sediment, followed by the pools present in the dissolved phase (DIC and DOC) and in biota. The sediment content of carbon is around 20 times larger than that in the other pools.

Confidence

Generally, the site descriptive model for the surface system is based on a wealth of site data. Uncertainties associated with the sub-models have been thoroughly evaluated, and descriptions and model results are, in most cases, consistent with regional/generic data and/or with the results from alternative models. The principal remaining uncertainties are associated with the description of the spatial distribution of the thickness of the overburden and the hydraulic properties of the shallow bedrock, especially outside the target area. The hydraulic description of the shallow bedrock is part of the hydrogeological bedrock model and there is substantial support for the existence of highly transmissive and connected structures in this part of the bedrock. Even though the general groundwater flow pattern is considered to be well known, it is difficult to establish the exact location of discharge areas of deep groundwater due to the poorly constrained geological characteristics and heterogeneous hydraulic properties of the sub-horizontal or gently dipping structures that obviously govern groundwater flow in the near-surface bedrock. Other remaining uncertainties concern the chemical composition of biota, the spatial variation of chemistry in the regolith, as well as the impact of chemical processes on transport of elements and quantitative estimates of processes, such as plant uptake and respiration. Most of these remaining uncertainties are judged to be of relatively minor importance for long-term safety or repository engineering, but uncertainties in the impact of chemical processes on the transport of elements contribute significantly to uncertainties in the SR-Site assessment of doses to humans and to the environment (cf. Section 13.2).

4.10.3 Human population and land use

The Forsmark region was not permanently settled until the end of Sweden’s prehistoric period (1100 AD). During the medieval period (1100–1550 AD), the region was characterised by small villages and new settlements were created in the areas peripheral to the older ones. At the end of the medieval period, the majority of the farms in the region belonged to freeholders, and only a few farms belonged to the church or the nobility. During the early modern period (1550–1750 AD), the establishment of the iron industry in the Forsmark region dramatically affected the surrounding landscape. Production was geared towards the needs of this industry; charcoal production, mining and the production of fodder for animals used in the industry. There was a strong population expansion and many crofts were established in the forested areas, inhabited by people involved in the production of charcoal.

During the 18th century, the number of freehold farms decreased, both due to the partitioning of farms and to the fact that large estates expanded. The population increased dramatically up to the late 19th century. At the turn of the century the increase ceased and, during the latter part of the 20th century, the rural population decreased. The number of people involved in agriculture decreased and, in contrast, the number of people employed in industry and crafts increased.

At the present day, the Forsmark parish is sparsely populated and the Forsmark model area has no permanent inhabitants. However, the nuclear power plant, situated immediately northwest of the planned repository, is a large industry with around 1,000 employees, which leave its mark on the area today. Besides this, there is a small holiday population manifested by the presence of five holiday cottages. The land use in the parish is dominated by forestry, and wood extraction is the only significant outflow of biomass from the area. The agriculture in the parish is limited in extent and there is only one agricultural enterprise in operation within the Forsmark area, situated at Storskäret.

5 Initial state of the repository

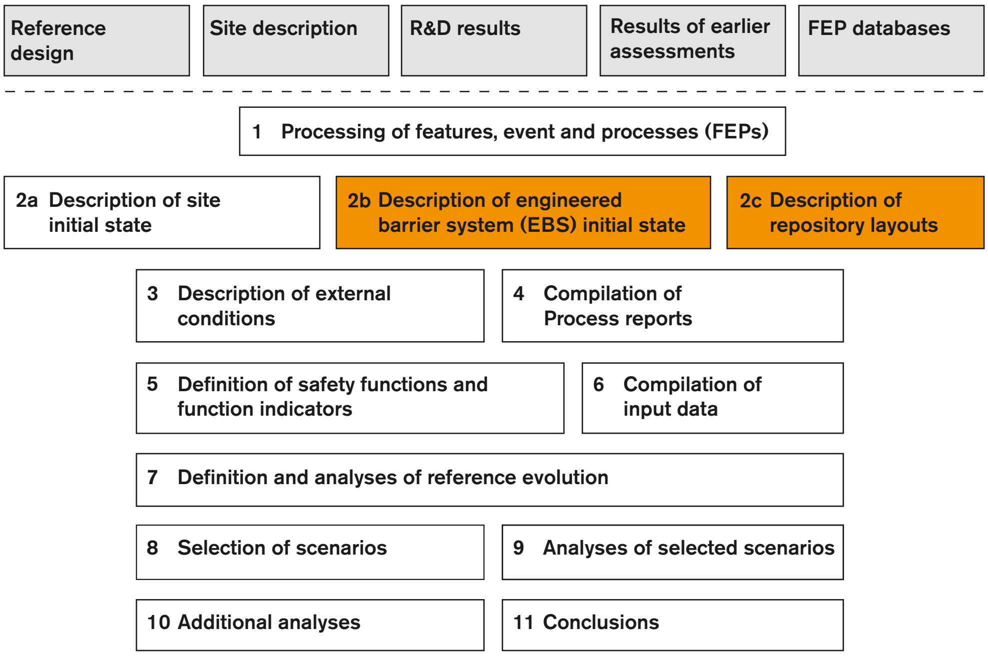


Figure 5-1. *The SR-Site methodology in eleven steps (Section 2.5), with the present step highlighted.*

5.1 Introduction

As mentioned in Section 2.5, a comprehensive description of the initial state of the repository system is one of the main bases for the safety assessment.

There is no obvious definition of the time of the initial state. For the geosphere and the biosphere, the state at the time of beginning of excavation of the repository is a natural starting point, since knowledge of this relatively undisturbed state is available through the site descriptive model that is derived from site investigation data. An alternative would be to consider, for each deposition hole, the state of the surrounding host rock at the time of deposition. Irrespective of which alternative is chosen, the short-term evolution of the host rock from the undisturbed state to that after excavation has to be considered in a safety assessment that is based on observations made prior to excavation. For the biosphere, the problem is less pronounced, since it will be less affected by the excavation of the repository.

For the engineered barrier system, the time of deposition/installation is a natural starting point when a specific part of the system is concerned, e.g. an individual deposition hole with its canister and buffer. However, if the entire ensemble of deposition holes is considered, there is no unique time of deposition. Neither is the time of repository closure a suitable choice for the engineered barrier system, since different parts of the repository will, at that time, have reached different stages of e.g. thermal and hydraulic evolution depending on the time of deposition and on spatial variability of rock conditions within the repository. The most reasonable approach is, therefore, judged to be to define the time of the initial state as that of deposition for each deposition hole with its canister, buffer and backfill, and then to describe the common evolution that all deposition holes will go through, taking spatial variability into account. For some aspects of the evolution, e.g. the thermal development, the deposition sequence has to be considered.

Based on these considerations, the initial state in SR-Site is defined as the state at the time of deposition/installation for the engineered barrier system and the natural, undisturbed state at the time of beginning of excavation of the repository for the geosphere and the biosphere. The evolution of the natural system is, therefore, at least in some aspects, followed from the time of beginning of excavation in the safety assessment. Short-term geosphere processes/alterations due to repository excavation are, therefore, documented in the **Geosphere process report**. An integrated description of the evolution driven by these processes is given in the excavation/operation phases of the reference evolution, Section 10.2.

The initial state of the engineered parts of the repository system is largely obtained from the design specifications of the repository, including allowed tolerances or deviations. Also the manufacturing, excavation and control methods have had to be described in order to adequately discuss and handle hypothetical initial states outside the allowed limits in the design specifications. The initial state of the engineered parts of the repository system for SR-Site is compiled in a number of dedicated **Production reports**, see further Section 5.1.1.

The initial state of the geosphere and the biosphere is, as mentioned, determined by site investigations. Field data from the site investigations are analysed, within the site investigation project, to produce a site descriptive model of the geosphere and the biosphere for the Forsmark site, as reported in the **Site description Forsmark** and summarised in Chapter 4.

This chapter contains a description of the initial state with uncertainties, summarising information on the engineered components and the repository layout from the **Production reports**, and the results of the FEP analyses reported in the **FEP report**. The level of detail in this chapter is meant to be sufficient for understanding the remaining parts of the safety report without reading the above reference documents.

5.1.1 Relation to Design premises, Production reports and Data report

Feedback from assessments of long-term safety is a key input to the refinement of the design of the KBS-3 repository. Feedback on the design was given in the SR-Can main report, Section 13.4. This was further developed into requirements termed *design premises* in a report entitled "Design premises for a KBS-3V repository based on results from the safety assessment SR-Can and some subsequent analyses", /SKB 2009a/. Design premises typically concern specification on what mechanical loads the barriers must be able to withstand, restrictions on the composition of barrier materials or acceptance criteria for the various underground excavations. The following approach was used.

- The reference design analysed in the SR-Can assessment was a starting point for setting safety related design premises for the next design step.
- A few design basis cases, in accordance with the definition used in the regulation SSMFS 2008:21 and mainly related to the canister, could be derived from the results of the SR-Can assessment. From these it was possible to formulate some specific design premises for the canister.
- The design basis cases involve several assumptions on the state of other barriers. These implied conditions were thus set as design premises for these barriers.
- Sections 13.5 and 13.6 of the SR-Can main report provide substantial safety related feedback on most aspects of the analysed reference design. Relevant parts of this feedback were also formulated as design premises.
- The safety function indicator criteria defined in SR-Can (see further Sections 8.3 and 8.4 of the present report) were also used as a basis for formulating design premises, noting that they are not the same as the design premises. Whereas the former indicate when a particular function is no longer sufficiently upheld, the latter refer to the initial state and must be defined such that they give a margin for deterioration over the assessment period. The basic approach for prescribing such margins is to consider whether the design assessed in SR-Can was sufficient to result in safety. In cases where this design would imply requirements that were too strict, and in cases where the SR-Can design was judged inadequate or not sufficiently analysed in the SR-Can report, additional analyses were undertaken to provide a better basis for setting the design premises.

The described methodology was applied and resulted in close to 30 different design premises on the canister, the buffer, the deposition holes, the deposition tunnels and backfill and on the main tunnels, transport tunnels, access tunnels, shafts, central area and closure. The resulting design premises constitute design constraints, which, if all fulfilled, form a good basis for demonstrating repository safety, according to the analyses in SR-Can and subsequent analyses.

This feedback has been a key input to the latest refinement cycle of the KBS-3 repository design.

The information required for the SR-Site assessment of engineered components of the repository system is described in a number of **Production reports** covering the spent fuel, the canister, the buffer, the tunnel backfill, the repository closure and the underground openings.

Each report gives an account of i) the design premises to be fulfilled, ii) the reference design selected to achieve the requirements, iii) verifying analyses that the reference design does fulfil the design premises, iv) the production and control procedures selected to achieve the reference design, v) verifying analyses that these procedures do achieve the reference design and vi) an account of the achieved initial state. The last is the key input to the safety assessment.

It is possible that the selected reference design and the production and control procedures yield an assessed initial state more favourable for long-term safety than one that just fulfils the design premises. In SR-Site, credit is generally not taken for this, "better-than-demanded" performance although it is the expected outcome of the production of the actual reference design that is assessed. The only exception is the requirement on the canister's resilience to isostatic load, where the pessimistically assessed highest future load (Section 12.7) somewhat exceeds that corresponding to the design premise, but where a canister insert produced according to the reference design is nevertheless assessed to withstand this load.

Furthermore, if the reference design is significantly better than the requirements expressed by the design premises, it is argued that this may form the basis for future revision of the design premises. Generally, the design premises may be modified in future stages of SKB's programme. Reasons for such modifications include results of analyses based on more detailed site data and a more developed understanding of processes of importance for long-term safety. Feedback on current design premises is given in Section 15.4.

The initial state, as given in the **Production reports**, provides quantitative information on key inputs to the safety assessment. These are critically evaluated in the **Data report** where the formal qualification of input data to the safety assessment occurs based on an evaluation of uncertainties affecting the initial state data.

5.1.2 Overview of system

The repository system is based on the KBS-3 method, in which corrosion resistant copper canisters with a load-bearing cast iron insert containing spent nuclear fuel are surrounded by bentonite clay preventing groundwater flow and deposited at approximately 500 m depth in groundwater saturated, granitic rock, see Figure 5-2.

The facility design with rock caverns, tunnels, deposition positions etc. is based on the design originally presented in the KBS-3 report /SKBF/KBS 1983/ which has since been developed and described in more detail. The deposition tunnels are linked by main tunnels for transport and communication. One ramp and several shafts connect the surface facility to the underground repository. The ramp is used for heavy and bulky transports and the shafts are used for utility systems, ventilation and for transport of excavated rock, backfill and staff. The different parts of the final repository are outlined in Figure 5-3.

Around 54,000 spent fuel assemblies corresponding to around 12,000 tonnes (heavy metal – initial weight) of spent nuclear fuel are forecast to arise from the Swedish nuclear power programme (see the **Spent fuel report**), corresponding to roughly 6,000 canisters in the repository. These figures are based on assumed reactor operational times of 50–60 years. The SR-Site assessment is, therefore, based on a repository with 6,000 canisters, corresponding to around 12,000 tonnes of fuel.

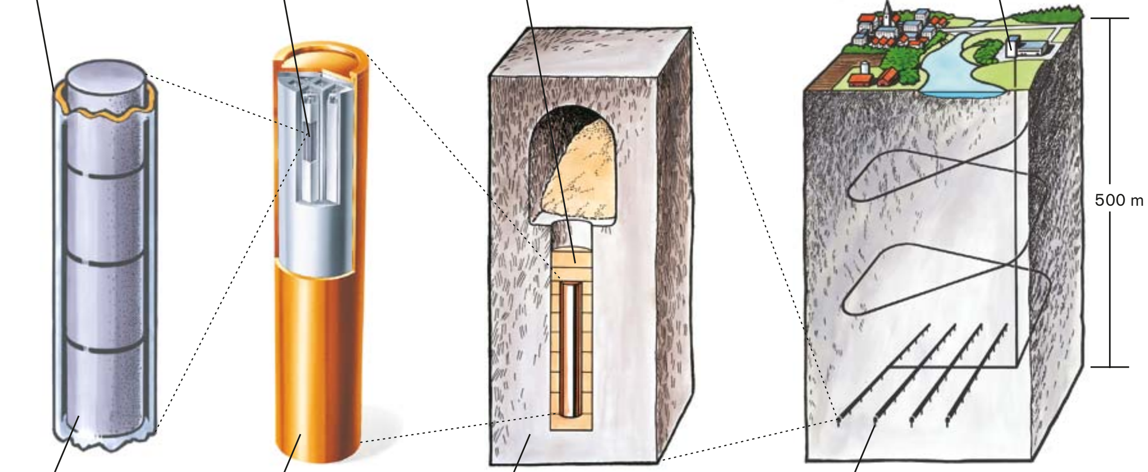


Figure 5-2. The KBS-3 concept for storage of spent nuclear fuel.

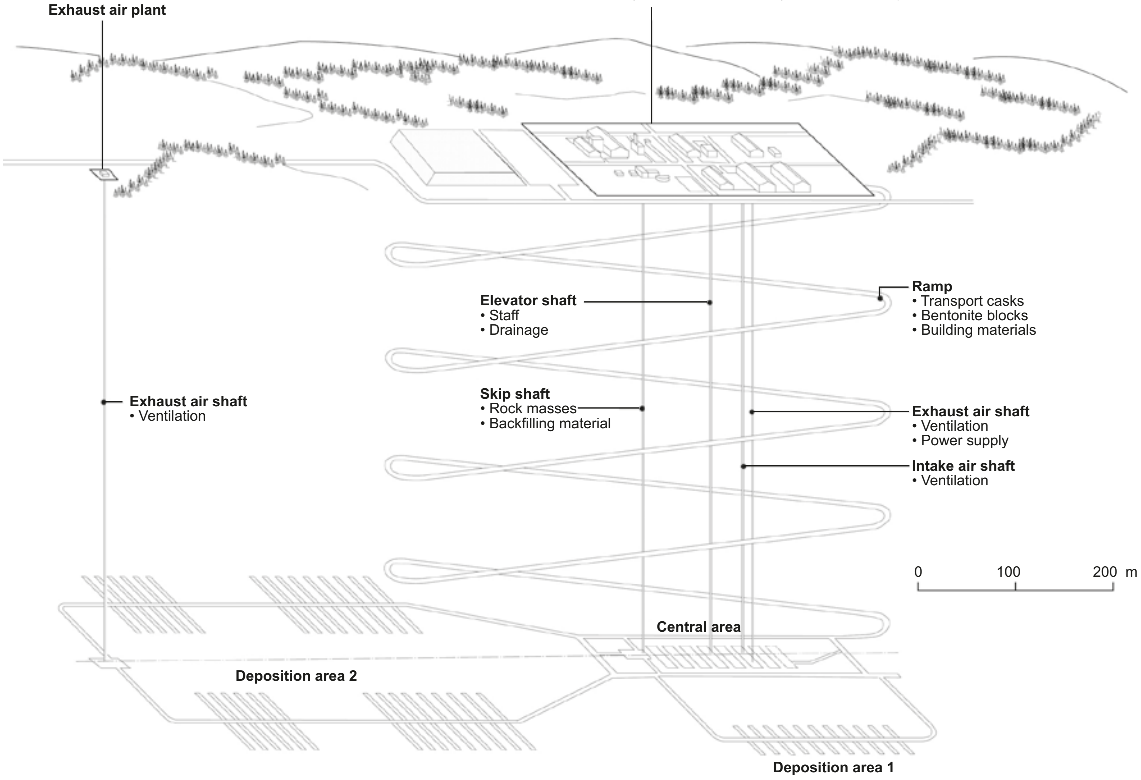


Figure 5-3. General repository layout showing the location of the underground functional areas (Access, Central and Deposition areas) and the surface facilities.

For the purposes of the safety assessment, the final repository system has been sub-divided into a number of components or sub-systems. These are:

- The host rock, see Chapter 4.
- The biosphere, see Chapter 4.
- The site adapted repository layout including descriptions of all underground openings, i.e. deposition holes, deposition tunnels, transport tunnels, the central underground area and access shafts and ramp, see Section 5.2.
- The fuel, (also including cavities in the canister since strong interactions between the two occur if the canister is ruptured), see Section 5.3.
- The cast iron insert and the copper canister, see Section 5.4.
- The buffer in the deposition hole, see Section 5.5.
- The backfill material in the deposition tunnel, see Section 5.6.
- Other engineered parts of the repository, see Section 5.7:
  - The closure backfill materials in transport tunnels, the central underground area, shafts and ramp.
  - The bottom plate in the deposition hole.
  - Plugs.
  - Investigation boreholes with their means of sealing.

This particular sub-division is dictated by the desire to define components that are as homogeneous as possible without introducing an unmanageable multitude of components. Homogeneity facilitates both characterisation of a component and the structuring and handling of processes relevant to its long-term evolution. Also, the importance of a particular feature for safety has influenced the resolution into components. In principle, components close to the potential source term, i.e. the spent fuel, and those that play an important role for safety are treated in more detail than peripheral components.

5.1.3 Initial state FEPs

As mentioned in Chapter 3, initial state FEPs in the SR-Site FEP catalogue are either related to an initial state in conformity to the specification given for the reference design or to deviations from the reference design. The former of these are handled in the category of variables in the SR-Site FEP catalogue. In these variable records, a reference is given to the description in the appropriate **Production report** of the reference initial state for that variable.

The initial state FEPs in the SR-Site FEP catalogue that are related to deviations from the reference design of the canister, the buffer and the backfill of the deposition tunnels, or to more general deviations, are compiled in Table 5-1. One such FEP of more general character is related to severe mishaps not expected to occur during the operational phase, like fires, explosions, sabotage and severe flooding or other events occurring prior to closure. Such events are excluded from the scenario selection. The reasons for this are i) the probabilities for such events are low and ii) if they occur, they shall be reported to SSM, their consequences assessed and correcting or mitigating actions made accordingly.

Another FEP in the SR-Site FEP catalogue refers the effects of phased operation. This affects mainly the geosphere and the subsequent development of the entire repository. The hydrological state of the bedrock is perturbed as soon as repository excavation starts (a smaller perturbation even occurs earlier during site investigations). Different parts of the repository, completed at different times, will be exposed to different hydrological conditions, affecting e.g. the saturation of the buffer and backfill. Possible upconing of saline water could also vary between different parts of the repository due to phased operation. Other factors to consider are the effects of blasting and underground traffic on completed parts of the repository. All these issues are part of the expected evolution of the repository, but are not automatically captured in the system of processes describing the repository evolution over time or by the initial state descriptions. As they need to be adequately included in the discussion of the repository evolution, they are propagated to the analysis of the reference evolution in Section 10.2.6.

Other FEPs in the FEP catalogue concern the effects of an abandoned, not completely sealed repository or open monitoring boreholes or shafts. These issues are also propagated to the scenario selection in Chapter 11.

FEPs relating to effects detrimental for long-term safety caused by monitoring are excluded from further analysis since monitoring activities that could disturb the repository safety functions will not be accepted.

Several FEPs concern design deviations due to undetected mishaps during manufacturing, transportation, deposition and repository operations etc. Measures to avoid or mitigate such deviations during excavation, manufacturing, handling, deposition etc. are described in the **Production reports**. To the extent that such mishaps may still occur, these issues are addressed in the scenario selection and the scenario analyses described in Chapters 11 and 12, respectively. Only such FEPs defined for the canister, buffer and the backfill of repository tunnels are included in Table 5-1. In addition to these FEPs, the FEP catalogue contains corresponding initial state FEPs for the remaining system components, i.e. the bottom plate in the deposition holes, plugs, borehole seals and closure in the central area, ramp shafts and tunnels other than deposition tunnels. Since these system components are not of primary importance for the safety of the repository, no safety functions have been assigned to them. Therefore, the consequences of deviations in their initial state are not analysed in detail, but are addressed in the analysis of the reference evolution (Chapter 10) and, if relevant, considered in subsequent parts of the assessment.

Table 5-1 Initial state FEPs in the SR-Site FEP catalogue and how they are handled in SR-Site.

Initial state FEP		Handling in SR-Site	FEP chart item (see Section 8.5)	Comment
ISGen1	Major mishaps/ accidents/sabotage	Excluded. The probabilities for such events are low. If they occur, this will be known prior to repository sealing so mitigation measures and assessments of possible effects on long-term safety can be based on the specific real event.		
ISGen2	Effects of phased operation	Assessed based on thermal, rock mechanics and transient hydrogeological simulations for an open repository.		See Section 10.2.6
ISGen3	Incomplete closure	Considered in scenario selection.		See Section 11.2 (scenario selection) and 14.2 (scenario analysis)
ISGen4	Monitoring activities	Excluded. Monitoring activities that could disturb the repository safety functions will not be accepted.		
ISC1	Mishaps – canister	Considered in the selection of scenarios based on safety function indicators related to canister integrity.	Copper thickness Canister design analysis	See Sections 11.2 (scenario selection) and 12.6 to 12.8 (scenario analyses).
ISC2	Design deviations – canister	Considered in the selection of scenarios based on safety function indicators related to canister integrity.	Copper thickness Canister design analysis	See Sections 11.2 (scenario selection) and 12.6 to 12.8 (scenario analyses).
ISBu1	Mishaps – buffer	Considered in the selection of scenarios based on safety function indicators related to buffer performance.	Density Geometry	See Sections 11.2 (scenario selection) and 12.2 to 12.4 (scenario analyses).
ISBu2	Design deviations – buffer	Considered in the selection of scenarios based on safety function indicators related to buffer performance.	Density Geometry	See Sections 11.2 (scenario selection) and 12.2 to 12.4 (scenario analyses).
ISBfT1	Mishaps – backfill in tunnels	Considered in the selection of scenarios based on safety function indicators.	Density Geometry	See Sections 11.2 (scenario selection) and 12.2 (scenario analyses). Transport properties of defective backfill addressed in Section 13.7.
ISBfT2	Design deviations – backfill in tunnels	Considered in the selection of scenarios based on safety function indicators.	Density Geometry	See Sections 11.2 (scenario selection) and 12.2 (scenario analyses). Transport properties of defective backfill addressed in Section 13.7.

5.2 Site adapted repository – the underground openings

The underground openings are the cavities constructed in the rock that are required to accommodate the sub-surface part of the final repository facility. The underground openings comprise:

- The actual geometry and location of the excavations.
- The rock surrounding the openings that is affected by the rock construction works.
- Engineered materials for sealing and rock reinforcement, and residual materials from performance of activities in the final repository facility which, at deposition, backfilling or closure, remain in and on the rock that surrounds the openings.

The underground openings as such do not contribute to the safety of the KBS-3 repository and do not have any barrier functions. However, the locations of the deposition areas and deposition holes with respect to the thermal, hydrological, mechanical and chemical properties of the rock are important for the utilization of the rock as a barrier and thus for the safety of the repository. Furthermore, the potential Excavation Damaged Zone (EDZ) and engineered and stray materials that remain in the rock may impact the barrier functions of the rock and/or the engineered barriers, and must therefore be known when assessing the safety of the repository.

The reference design is described in the **Underground openings construction report**, Chapter 4, but a more detailed description and justification of the design is provided in /SKB 2009b/. The basis for this reference design are site-specific geotechnical information which has been interpreted and evaluated in a Site engineering report (SER) /SKB 2009c/, building on the extensive surface-based site investigations and their evaluation and modelling presented in the **Site description Forsmark**, and the stated design premises with respect to long term safety and other design premises, as specified in the **Underground openings construction report**. A part of the design work has been to assess the risk of practically achieving this design in relation to the current confidence and remaining uncertainty in the **Site description Forsmark**, see Chapter 8 of /SKB 2009b/. This risk assessment concluded that none of the consequences from these uncertainties would render the repository unsuitable for the purpose intended. However, several uncertainties were identified that should be resolved during the next design step and/or during construction of Repository Access.

Similar to the other construction reports, see Section 5.1.1, the design is based on design premises provided from a long term safety perspective /SKB 2009a/, from design decisions affecting different parts of the repository facility and from some more general principles /SKB 2007b/ stated by SKB.

5.2.1 Design premises relating to long-term safety

The following design premises /SKB 2009a/ apply to the underground openings:

Selecting repository depth and repository areas: The repository volumes and depth need to be selected where it is possible to find large volumes of rock fulfilling the specific requirements on deposition holes. The minimum depth is prescribed to be as specified for a KBS-3 repository i.e. at least 400 m.

Adapted to the chemical conditions at the site[6]: The groundwater composition in rock volumes selected for deposition holes should, prior to excavation, fulfil the SR-Can function indicator criteria regarding chemically favourable conditions: These criteria are reducing conditions; salinity in terms of total dissolved solids (TDS) should be limited, ionic strength $[M^{2+}] > 1$ mM, concentrations of K, HS^-, Fe limited, pH < 11, pH > 4 and $[Cl^-] < 2$ M. When quantitative criteria are not given, the term "limited" is used to indicate favourable values of the safety function indicators.

[6] These design premises are likely to be revised later since SR-Site has revised the related safety function indictors, see Section 8.3.4.

Mitigating earthquake hazard: Deposition holes are not allowed to be placed closer than 100 m to deformation zones with trace length longer than 3 km. Deposition holes should be, as far as reasonably possible, selected such that they do not have potential for shear larger than the canister can withstand. To achieve this, the "Extended Full Perimeter Criterion (EFPC)", see Section 5.2.2 for a definition, should be applied in selecting deposition hole positions.

Adapted to the hydrological and transport conditions at the site: The total volume of water flowing into a deposition hole, for the time between when the buffer is exposed to inflowing water and saturation, should be limited to ensure that no more than 100 kg of the initially deposited buffer material is lost due to piping/erosion. This implies, according to present knowledge, that this total volume of water flowing into an accepted deposition hole must be less than 150 m^3. Fractures intersecting the deposition holes should have a sufficiently low connected transmissivity (though a specific value cannot be given at this time). This condition is fulfilled if the conditions regarding inflow to deposition holes are fulfilled.

Adapted to the thermal conditions at the site: Buffer geometry (e.g. void spaces), water content and distances between deposition holes should be selected such that the temperature in the buffer never exceeds 100°C. (The maximum thermal output from the canister is limited to 1,700 W).

Restrictions on deposition hole wall transmissivity: Before canister emplacement, the connected effective transmissivity integrated along the full length of the deposition hole wall and as averaged around the hole, must be less than 10^{-10} m^2/s.

Excavation damaged zone (EDZ) in deposition tunnels: Excavation induced damage should be limited and not result in a connected effective transmissivity, along a significant part (i.e. at least 20–30 m) of the disposal tunnel and averaged across the tunnel floor, higher than 10^{-8} m^2/s.

EDZ in shafts and ramp, rock caverns and tunnels other than deposition tunnels: Below the location of the top sealing[7], the integrated effective connected hydraulic conductivity of the backfill in tunnels, ramp and shafts and the EDZ surrounding them must be less than 10^{-8} m/s. This value need not be upheld in sections where e.g. the tunnel or ramp passes highly transmissive zones. There is no restriction on the hydraulic conductivity in the central area.

Grouting and reinforcement in deposition tunnels: Only low pH[8] materials (pH < 11) allowed. No continuous shotcrete. A continuous array of grouting boreholes outside the tunnel perimeter should be avoided.

Grouting in boreholes shafts and ramp, rock caverns and tunnels other than deposition tunnels. Only low pH (< 11) materials are allowed below the level of the top seal.

The justification of the design premises is found in the design premises report /SKB 2009a/. As noted in the introduction, Section 5.1, the design of the underground openings as well as the as-built and inspected underground openings shall conform to the design premises.

5.2.2 Repository design and resulting layout

The design premises for the different underground openings in the final repository facility based on the results from the long-term safety assessment are presented above. The design premises establish the acceptable layout and properties of the underground openings. The objective of rock engineering is to ensure that the layout of the repository facility:

- Is well adapted to the site conditions.
- That it conforms to the design premises in any phase of design and construction as well as after the construction is completed.

[7] See e.g. Figure 5-25.

[8] "Low" refers to conditions to be expected in cementitious environments. Generally pH less than 11 is not considered "low".

Design stages

In all phases of underground design, uncertainties with regard to site conditions must be anticipated. In order to establish a final layout for deposition tunnels and deposition holes, a large volume of rock will have to be characterised, but this characterisation can only effectively be carried out from the existing underground openings. This means that the characterisation will develop as the construction work proceeds.

The primary uncertainties that will influence the final layout are the spatial location and variability of the geological setting and the potential behaviour of the rock mass with respect to excavation also considering the effects of rock support and grouting measures. These uncertainties and the scale of the repository volume emphasize that the methodology used to adapt the final layout of the repository to the site conditions must be integrated with the construction activities required to develop the repository.

The development of a final layout for the repository facility based on site adaptation requires the use of an iterative design process based on the *Observational Method*. The Observational Method is a risk-based approach to underground design and construction that employs adaptive management, including advanced monitoring and measurement techniques, see further Chapter 3 of the **Underground openings construction report**.

The design assessed in SR-Site is preliminary. As more knowledge is attained during the excavation and connected detailed characterisation, more refined designs will be developed. Rock engineering for the underground openings in the final repository facility will therefore be divided into several stages prior to construction as described in Chapter 3 of the **Underground openings construction report**. In this context, the current Reference Design D2 /SKB 2009b/ documents the results from a preliminary design stage which established the feasibility of the repository facility. In later stages, when the design is developed into the detailed and final stages, relevant parts of the safety assessment may need to be updated. This is further discussed in Chapter 15.

It is foreseen that there will only be minor changes in the actual constructed layout compared to the layout developed in detailed design and that site adaptation will primarily concern rock support, grouting activities and the final positioning of deposition tunnels and deposition holes. The latter decision will require a characterisation of the rock mass on location, something which is a major topic for the investigation, inspection and monitoring programme. Finally, verifying that the drilled deposition holes conform with all design premises such that they can be used as intended may require additional input from monitoring and inspections carried out during the period between completion of excavation and deposition.

Repository depth

The depth established for the reference design is a compromise arising from design premises on long-term safety and constructability of the deposition tunnels and deposition holes of the repository facility. According to the design premises the depth should be selected "where it is possible to find large volumes of rock fulfilling the specific requirements on deposition holes" and the premises also state a minimum depth of 400 m. A rationale for identifying suitable rock volumes for deposition as well as depth intervals for the final repository facility has been outlined in the SER /SKB 2009c/. This rationale has been used to establish a depth interval where it is possible to find rock volumes that fulfil the specific requirements for deposition holes and deposition tunnels with regard to:

- Available space.
- Fracture frequency and frequency of connected transmissive fractures.
- Groundwater pressure.
- *In situ* stress magnitudes and orientation.
- Initial temperature and thermal properties of the rock.
- Salinity and up-coning.
- Chemical conditions.
- Lengths and transport resistances of hydraulic travel paths to and from the repository.

These factors are assessed in the SER /SKB 2009c/ using the information in the **Site description Forsmark**. At Forsmark, it is mainly the hydraulic conditions of the site, i.e. frequency and occurrence of transmissive fractures and its dependency on depth that are of importance for safety, while the constructability is mainly related to rock mechanics issues, e.g. the likelihood and extent of spalling in deposition holes prior to emplacement. Below the minimum depth of 400 m it is found that the main conditions of importance are the frequency of water conducting fractures and the *in situ* stress magnitude, whereas other factors show little or moderate change with depth. While the frequency of water conducting fractures is already low at depth, this frequency dramatically drops at depths greater than 400 m pointing to a potential major advantage for depths below 450 m. Below 300 m depth, there appears to be little evidence that the horizontal stress magnitudes in fracture domain FFM01 increase significantly with depth. Hence placing the repository at 400 m or 500 m depth does not significantly increase the risk for excavation-induced spalling in the deposition holes. For these reasons, the SER suggests locating the repository at Forsmark at a depth range of between 450 m and 500 m.

In the preliminary design developed, the maximum depth of the repository facility is located at elevation −470 metres, i.e. where the transport tunnels (tunnel floor) exit from the central area. The minimum depth (tunnel roof) of the reference design is located at elevation −457 metres, i.e. in some deposition tunnels. The respective locations of minimum and maximum depth for the repository facility will be based on the drainage system requirements.

Thermal dimensioning – distance between deposition holes

The basis of the thermal dimensioning for the reference design are: Fixed canister spacing, maximum thermal power of a canister 1,700 W, a minimum depositional hole spacing of 6 m, deposition tunnel spacing 40 m and a maximum allowed peak temperature in the buffer < 100°C. The latter is in accordance with the design premises that the distance between deposition holes must be sufficiently large to achieve a temperature in the buffer < 100°C.

The design methodology to meet this requirement is presented in /Hökmark et al. 2009/. The minimum spacing between deposition holes for the reference design at Forsmark was evaluated in the SER and the analysis gave provision for a minimum centre-to-centre spacing for deposition holes equal to 6.0 m in rock domain RFM029 and 6.8 m in rock domain RFM045.

Site adaptation with respect to deformation zones

The layout is adapted to meet the design premises relating to mitigating earthquake hazard, see Section 5.2.1. Within the target volume there are only four deformation zones that are large enough to potentially require a respect distance: the three steeply dipping zones ZFMENE060A, ZFMENE062 and ZFMNW0123, and the gently dipping zone ZFMA2, see Chapter 4.

Furthermore, large fractures are not allowed to intersect deposition holes in accordance with the Extended Full Perimeter Intersection Criterion (EFPC). The EFPC was originally presented by /Munier 2006/. However, in assessing the consequences of this criterion it has been found that it rejects many deposition positions that in fact are only intersected by relatively short fractures. In order to enhance the effectiveness of the criterion it is now reformulated in the following way, see also Figure 5-4:

- Deposition positions being intersected by a fracture that intersects the full tunnel perimeter and that also is projected to intersect the canister location in the deposition hole, are rejected (FPC).
- Deposition positions intersected by a fracture intersecting four or more additional potential deposition positions are rejected.

However, for practical reasons, the old definition, i.e. rejecting deposition holes intersected by a fracture also intersecting the full tunnel perimeter has not been altered for the far less frequent potentially water bearing fractures. For these fractures EFPC is used as an indicator for high flows once the repository is sealed and saturated.

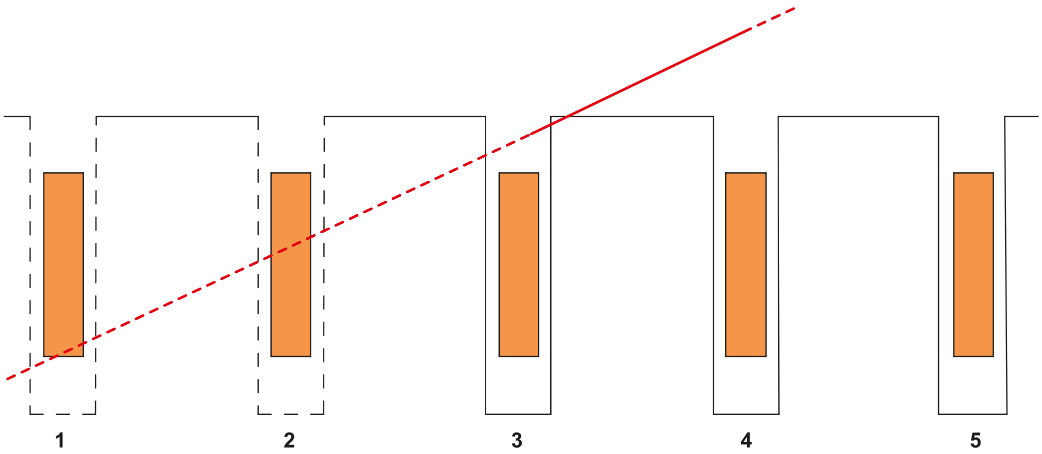


***Figure 5-4.** In SR-Site, the EFPC criterion is changed such that deposition positions being intersected by a fracture that intersects the full tunnel perimeter and that also is projected to intersect the canister location in the deposition hole, are rejected (i.e. positions 1 and 2). For potentially water bearing fractures the previous criterion still applies (i.e. position 3 is also rejected).*

In accordance with the criterion, deposition holes are not placed in positions intersected by deterministic deformation zones shorter than 3 km, even though deposition tunnels are allowed to intersect such zones. In addition to this, the anticipated loss of deposition hole positions due to application of the EFPC criterion defined above can currently only be assessed stochastically using the discrete fracture network model (DFN) of the **Site description Forsmark**.

The revised EFPC criterion was used when the reference design was established to assess the potential loss of deposition holes due to this criterion. According to the design report /SKB 2009b/ the loss of positions could range between 10 to 25% depending on which DFN model is used, but it is also stated that the actual loss of positions is judged much smaller, since prospects are good of finding more efficient means of identifying fractures that are too large.

Rock mechanics and rock support

General engineering guidelines, listed in the SER /SKB 2009c/, were considered in the reference design with regard to rock mechanics issues. These guidelines cover feasibility for both construction and deposition. Furthermore, the buffer and backfill imposes design premises on the acceptable geometry of the deposition holes and tunnels (see **Underground construction report**, Chapter 2). Potential alterations of the geometry from spalling as well as the capability of the drilling and excavation methods need to be considered when verifying the conformity to these design premises.

According to the design report /SKB 2009b/ the *in situ* stress conditions at Forsmark at the depth of the repository are not expected to be sufficient to cause extensive stress-induced stability problems in the form of spalling in the deposition tunnels, using the “most likely” stress model and for the current layout and tunnel orientations. However, there is uncertainty regarding this design parameter. Some evidence points to lower stress magnitudes while other evidence points to higher stress magnitudes. The evaluation of all possible stress models indicates that mitigation measures using reinforcement, tunnel orientation and opening shape should be enough to maintain the spalling at acceptable levels also for the access ramp, the central area, the main tunnels and the transport tunnels, even if these latter parts cannot always be aligned with the orientation of the maximum principal stress. The reference design includes quantities of rock support specifically aimed at reducing structurally-related overbreak to an acceptable level.

/Martin 2005/ showed that the likelihood of spalling in deposition tunnels could be significantly reduced – if not eliminated – by aligning the deposition tunnels parallel to the maximum horizontal stress. Such alignment would also reduce the potential for thermally induced spalling According to the guideline given in the SER the deposition tunnels shall be aligned within ±30 degrees of the trend of the maximum horizontal stress to significantly reduce the risk of spalling. A three dimensional elastic stress analysis, presented in the design report /SKB 2009b/, confirms this guideline.

For deposition holes, the three dimensional elastic analyses showed that for the "most likely" stress model, only the deposition tunnels aligned greater than 30 degrees to the maximum horizontal stress will produce tangential stress concentrations that are higher than the spalling strength, and in these situations the spalling will occur above the top of the canister. In the case of the "most likely" stress model, the results indicate that some 100–200 deposition holes (out of 6,000) would sustain a spalling depth (overbreak) that exceeds 5 cm, provided the deposition tunnels are aligned between 0 and 30 degrees to the maximum horizontal stress. For the "unlikely maximum" stress model, the deposition tunnel must be aligned parallel to the maximum horizontal stress, but the number of deposition holes that can sustain a spalling depth in excess of 5 cm is approximately the same, see Figure 5-5. Moreover, loose rock debris from localised spalling that remains on the rock walls in deposition holes can be scaled off to achieve compliance with the requirement for associated effective transmissivity.

In order to use space efficiently, the deposition tunnels in the reference design are aligned between azimuths 123 to 140 degrees. Should the "unlikely maximum" stress conditions be found when the excavation reached the repository level, these alignments must be adjusted.

In order to prevent minor spalling from being an operational safety issue, roof support with shotcrete is included in the reference design for all underground openings except in the deposition tunnels where wire mesh shall be used if necessary.

Some material in the rock or on rock surfaces e.g. originating from rock support will remain in the repository after closure. The assessed quantities are given in the **Underground openings construction report**, Table 4-1.

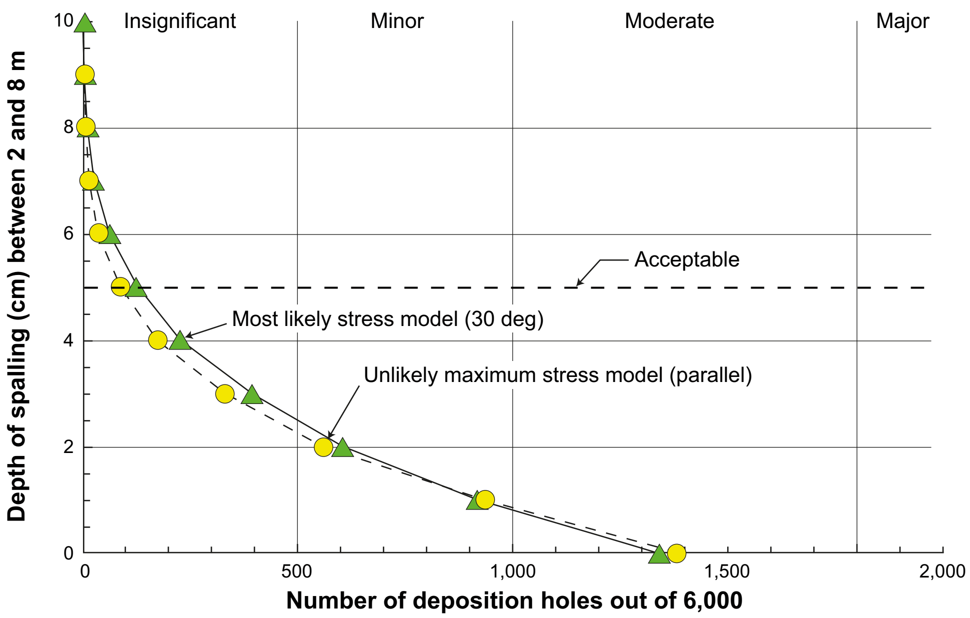


Figure 5-5. *Estimated depth of spalling in deposition holes for the "most likely" and the "unlikely maximum" stress models. For the "most likely" case deposition tunnels are assumed to be aligned between 0 and 30 degrees to the maximum horizontal stress, whereas for the "unlikely maximum case" deposition tunnels are assumed exactly parallel to the maximum horizontal stress (Figure 8-4 in /SKB 2009b/). The classification of number of holes affected being "insignificant", "minor", "moderate" and "major" relates to the design risk assessment presented in /SKB 2009b/.*

Adaptation to hydrogeological conditions

Several design premises concern limitations on inflow of water to different parts of the repository facility. In the design this is handled by not accepting deposition holes with too high inflows, by layout adjustments of other parts and by means of grouting. Grouting measures need in turn follow the restrictions on grouting material and placement of grouting holes. The composition and the amounts of the materials used for ground support and for grouting are given in Tables 4-1 and 4-2 of the **Underground construction report**.

The need for grouting and a rationale for assessing the quantities of cement, silica sol and additives for the reference design, including both the total amount and the amount remaining in the rock, is presented in the design report /SKB 2009b, Section 7.3/. In fracture domain FFM02, located within the upper 100–200 m of the rock, the rock has a relatively high frequency of transmissive fractures and relatively extensive grouting measures would be needed in this part of the rock. Below 200 m the frequency of transmissive fractures is generally low and decreases with depth and the grouting can be carried out as selective pre-grouting, with probe hole investigations, when passing deformation zones and where discrete water-bearing fractures are encountered. Below a depth of approximately 400 m, the observed frequency of flowing features is very low and grouting of flowing fracture and zones will be localised and not result in continuous grouting holes outside the deposition tunnel perimeter. On average, less than 2% of the 20 m sections between deformation zones will require grouting /SKB 2009b, Section 2.5/.

When grouting is required at the repository level, it is anticipated that cement-based grouting will be adequate to achieve the required sealing efficiency. However, for some water-bearing fractures and deformation zones in deposition tunnels it may not be practical to use cement-based grouts, and in order to achieve the required sealing efficiency, options were included in the reference design for the use of new technologies such as silica sol, which has recently been tested at the Äspö HRL /Funehag 2008/.

Resulting layout

The major site adaptation issues for the reference design layout discussed above are:

- Repository depths.
- The distance between deposition holes based on thermal dimensioning.
- Respect distance relative to modelled locations of major deformation zones.
- The alignment of deposition tunnels in relation to the direction of maximum horizontal stress.
- The disqualification of unsuitable deposition hole positions.

Considering these issues, a reference design layout for Forsmark was developed, see Figure 5-6, with the aim of maximising the gross capacity relative to potential deposition positions. The reference layout includes provision for all deterministic deformation zones identified in the site descriptive model. In addition, there is a respect distance of 100 m for deformation zones with a trace length longer than 3 km. There are no deposition positions in any of these zones in the layout. Furthermore, there are no deposition positions intersecting known deterministic minor deformation zones. The gross capacity of the reference design layout is 7,818 deposition positions and allows for a 23% loss of deposition positions relative to the about 6,000 canisters that are planned for the final repository facility, i.e. more than is expected to be needed. The reference design layout for the final repository facility is shown in Figure 5-6.

The reference design does not include any spare areas for deposition, in addition to the allowable 23% loss of positions, but recognises the fact that utilising other potentially suitable rock volumes would offer additional deposition positions. The reference design also acknowledges that the justification of spare capacity would require additional site investigations and modelling.

Since there is a need to maintain segregation between construction and emplacement zones during operation, a stepwise methodology for the successive extension of deposition areas is envisaged. A tentative stepwise development of the repository facility is presented in /SKB 2009b/.

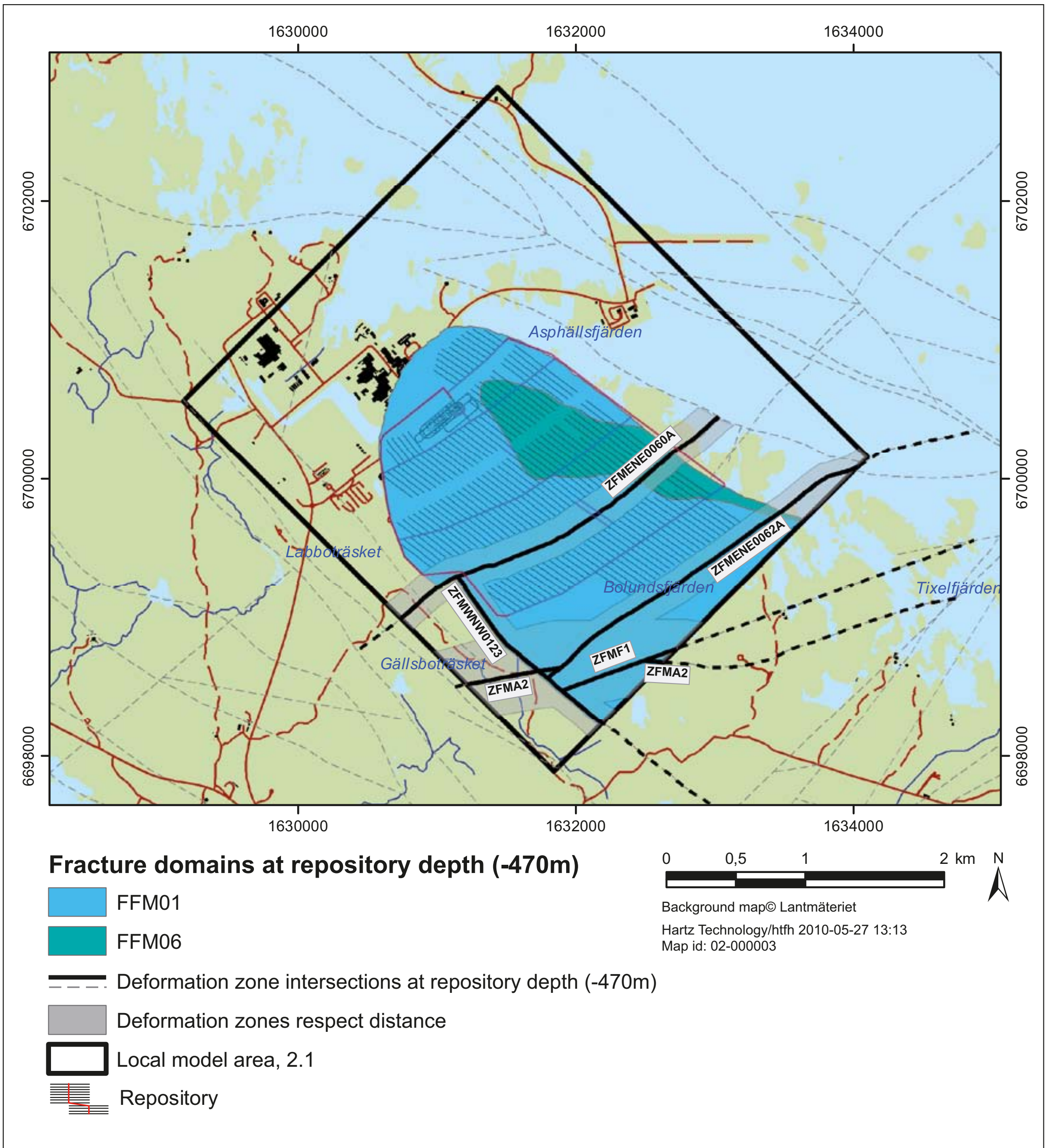


***Figure 5-6.** Reference design layout for the repository at Forsmark, also showing respect distances and the deformation zones requiring the respect distances (Figure 4-1 in the **Underground openings construction report**).*

5.2.3 Initial state of underground openings

Chapter 6 of the **Underground openings construction report**, is an assessment of the projected initial state of the openings at the time when deposition starts. This assessment is based on an evaluation of the adequacy of current design tools and planned excavation techniques together with an evaluation of the efficiency of potential control measures and contingency actions during and after construction. A formal assessment of the adequacy of these data are made in the SR-Site **Data report**; the findings given in this section are for information only.

Repository depth and repository areas

The repository design with its reference layout is justified and presented in the previous section. Assessment of the suitability of this layout is part of the SR-Site assessment and feedback to future stages of the repository programme is given in Sections 15.5 and 15.6 of this report.

It is judged that remaining uncertainties in the geological description can be sufficiently resolved using methods and techniques that were implemented during the site investigations and would only require minor re-adjustments of the available areas. Uncertainties in the orientation of maximum horizontal stress can only be significantly reduced by *in situ* tests at depth during access construction. The finding may necessitate a re-orientation of the deposition tunnels, but would not affect the overall suitability of the designated depth and repository areas.

Chemical conditions at the site

The site and layout of the repository is selected to conform to the stated design premises on the chemical conditions. Assessment of the actual chemical conditions, and more importantly their future evolution together with an assessment of the safety importance, is part of SR-Site and is presented in Chapter 10 of this report.

Thermal conditions

For the reference design, the minimum centre-to-centre spacing for the deposition tunnels is 40 m and the minimum centre-to-centre spacing for the deposition holes is 6 m in RFM029 and 6.8 m in RFM045. While there is no experience of heating large volumes of rock, the analytical and numerical techniques used to predict heat transfer are well established and the smaller scale experiments have validated the approaches used to verify the distance between deposition holes in the reference design /Hökmark et al. 2009/. There is high confidence in the thermal dimensioning methodology used for the reference layout and it is also recognised that there is a good potential for refining the design, once underground data are available, for further reducing the repository footprint. The adequacy of the current design has been re-assessed in SR-Site, using thermal data and other properties assessed in Sections 6.2 and 6.4 of the **Data report**, see Section 10.3.4.

An effective way of ensuring that the thermal properties are determined correctly is to combine geological mapping with measurement techniques. The measurement technologies and instruments utilised to determine thermal properties of the rock are well established. Details of the approach to developing tools for measuring and evaluating the thermal properties are presented in the programme for detailed investigation /SKB 2010b/.

Mitigating earthquake hazard

The reference design was developed in accordance with the stated design premises and considers all the deterministic deformation zones and respect distances for the deformation zones with a trace length exceeding 3 km. Deposition positions are not located in any of the deterministic zones, nor inside the given respect distances.

The location of discriminating fractures intersecting deposition holes cannot be determined deterministically at this stage, but the locations should be sufficiently known after deposition tunnels are excavated and characterised. Current plans consider application of the EFPC criterion, see Section 5.2.2.

The effective way of determining the extent of discriminating fractures and critical deformation zones in underground openings would be to combine geological mapping with geophysical techniques such as P-wave velocity and ground radar, see /Cosgrove et al. 2006/ and the **Underground openings construction report**, Section 5.3.4. The techniques already tested during the site investigations are judged essentially sufficient, but at this stage no specific method or combination of methods can be recommended. Hence the method for identifying and characterising discriminating fractures is conceptual and some uncertainties prevail in the reference design about the performance of the reference method relative to the EFPC criterion. Details of the programme to develop measurement tools for identifying and evaluating discriminating fractures intersecting deposition holes is presented in the programme for detailed investigation /SKB 2010b/.

In conclusion, it is judged sufficiently plausible that the EFPC can be applied successfully and that this can be assumed for in the SR-Site assessment. The impact of not applying the criterion is also illustrated in the assessment.

Hydraulic conditions around deposition holes

The design methodologies used to estimate groundwater inflow into underground excavations are well established in hydrogeology. Analytical and numerical methods have been used to estimate the inflows for the reference design. These numerical models were calibrated with respect to hydrogeology measurements for the site and hence there is confidence in the estimated inflows to the underground openings.

The total volume of water flowing into an accepted deposition hole, between buffer installation and the time when the buffer has saturated, must be less than 150 m^3. For the current reference design it is judged that the design premises are met if only potential deposition holes with inflows less than 0.1 L/min are accepted. The potential loss of deposition positions due to this criterion has been assessed in the Site engineering report /SKB 2009c/. In summary, the most likely situation is that very few additional deposition holes will be rejected due to high inflows since most of the few high flow positions are likely to be already screened out by the (modified) EFPC criterion. At the most extreme, i.e. assuming that this screening is not effective, an additional 6% could be rejected due observed inflows higher than 0.1 L/s.

It is relatively straightforward to monitor the inflow to deposition holes and to evaluate the initial state, see the **Underground openings construction report**, Section 5.3.3. Should the measured inflow not conform to the design premises the deposition hole will be rejected. Details of the programme to select measurement techniques for evaluating the inflows of water to deposition holes and how these flows will be measured and evaluated are presented in the programme for detailed investigation /SKB 2010b/.

The deposition hole selection criteria will also affect the groundwater flow conditions applicable to deposition holes after closure. This applies both to EFPC primarily introduced to mitigate the earthquake hazard and the omission of deposition holes with high inflows.

The successful application of the EFPC is also assumed for the hydrogeological simulations, i.e. deposition holes intersected by flowing fractures that fulfil the EFPC are omitted. However, in applying the EFPC for the hydrogeological analyses, based on the hydrogeological DFN model having a much lower fracture intensity than the geological DFN, deposition holes intersected by FPC fractures are omitted regardless of whether they intersect the canister or not, for details see Figure 5-4.

A transmissivity related rejection criterion is also assumed in SR-Site according to the following: An inflow of 0.1 L/min would be produced by a large fracture and with a transmissivity of $4 \cdot 10^{-9}$ m^2/s /Smith et al. 2008, Appendix B2/. However, considering that inflow may be reduced by e.g. intersecting fractures and local skin effects it is in SR-Site judged to be too optimistic to assume that all fractures of this transmissivity or higher would be found when applying the inflow criterion and thereby avoided when selecting deposition hole positions. It is, however, assessed as justified to assume that very large and very transmissive fractures would be detected. In SR-Site it is assumed that fractures, or rather minor deformation zones, with a radius larger than 250 m and with efficient transmissivity larger than 10^{-6} m^2/s, will be detected by the detailed investigations so that potential deposition positions intersecting such fractures would be avoided. Such fractures would have the potential of an inflow in the order of 25 L/min. Furthermore, they would also be easy to recognize by local geophysics and other tools applied within the detailed investigation programme. The size limit of 250 m is the same as that applied when assessing critical deposition positions regarding shear movements, see Section 10.4.5. The transmissivity limit of 10^{-6} m^2/s is seen as cautious considering the characteristics of such a fracture discussed above. In all hydrogeological calculations, this transmissivity/fracture length (T/L) criterion is implicitly included in the EFPC criterion, unless otherwise stated. (It is demonstrated in Section 14.3 that this additional T/L criterion has a much more limited impact than the 'pure' EFPC criterion.)

Furthermore, based on the assessment results of SR-Site, presented later in this report, the deposition hole selection criteria might be revised compared to those currently assessed, as discussed in Sections 14.3 and 15.5 of this report.

Deposition hole geometry

The results from full face down-hole drilling of 10 deposition holes at the Hard Rock Laboratory in Äspö were used to investigate the performance of the reference method for excavating deposition holes. The variability of the mean cross-section diameter relative to the mean diameter of the deposition hole is well within the current geometrical tolerances.

The effective way of ensuring that the geometrical tolerances are fulfilled would be to combine quality assurance in the execution of works with measuring techniques to determine the as-built dimensions. There are methods and instruments that are potentially fit for measuring the geometry of deposition holes after excavation, e.g. laser scanning and geodetic methods. However, the current methods are considered to be conceptual and although they are based on proven technology, a certain degree of new technology and innovation is likely to be required to develop a suitable field method, as discussed in Section 5.3.2 of the **Underground openings construction report**.

Deposition hole EDZ

Findings from a comprehensive literature study /Bäckblom 2009/ as well as the **Underground openings construction report**, Section 5.3.1, conclude that for mechanical excavation techniques in elastic rock conditions, i.e. full face down-hole drilling, the depth of the excavation damaged zone (EDZ) is limited to a few centimetres in the rock surrounding the deposition hole. The hydraulic conductivity in such a zone is in the order of 10^{-10} m/s or less. There is high confidence that elastic rock conditions prevail for the reference design and consequently that the transmissivity of the EDZ in deposition holes, if such a zone exists at all, would be less than 10^{-10} m^2/s.

The magnitude of the connected effective transmissivity may be altered due to occurrences of spalling. While there is high confidence in the design methodology utilised to assess the spalling potential for the reference design, there are uncertainties relating to the *in situ* stress conditions and the rock properties. This restricts the capability of modelling the extent of spalling and the associated change in transmissivity at this stage. It was assessed that approximately 100–200 deposition holes (out of the about 6,000) would be subject to overbreak that exceeds 5 cm, see Figure 5-5. Because the design methodology is empirically based, additional investigations will be needed at the repository level to confirm the design assumptions. Details of the programme to develop means for evaluating the stress at the repository level are presented in /SKB 2010b/.

Should spalling occur, loose rock debris from localised spalling on the rock walls would be removed. The ultimate contingency action is to reject the deposition hole. In conclusion, spalling prior to deposition would not affect the geometry of accepted deposition holes outside the stated margins.

Excavation Damaged Zone (EDZ) in deposition tunnels and other openings

The potential development of a damaged zone in underground openings excavated by drill and blast is discussed in Section 5.2.1 of the **Underground construction report**. /Bäckblom 2009/, a reference to the **Underground construction report**, states that many studies correlate damage with the concentration of explosives used. Equally important are the accuracy and precision in drilling the blast holes, and using exact ignition times, e.g. by utilising electronic detonators /Christiansson et al. 2009/. Local geological conditions and the rock stress environment will also influence the resulting excavation damage zone /Jonsson et al. 2009a/.

It is important whether the fractures induced by the excavation form a connected network, since if not there will be no conductive EDZ at all. Based on experience from the excavation of the TASQ tunnel at the Äspö Hard Rock Laboratory /Olsson et al. 2004/ it was noted in the SR-Can assessment that it is possible to design and control the drilling and blasting of tunnels such that continuous fracturing along the axial direction of the tunnel will not develop. This notion has now been demonstrated at the Äspö Hard Rock Laboratory in the TASS tunnel using smooth-blasting techniques /Olsson et al. 2009/ and /Ericsson et al. 2009/. Those results showed that proper control of drilling and blasting procedures resulted in blast-induced fractures that are dominantly radial in direction and that such fractures are not continuous along the axial direction of the tunnel over any significant distance.

If a connected zone at all develops, a reasonable value for the hydraulic conductivity of the damage zone is in the order of 10^{-8} m/s. This magnitude has been obtained during several tests in crystalline rocks, where excavation was of good quality and measured by integrating measurement under saturated conditions along the tunnel floor /Bäckblom 2009/. Point observations of the hydraulic conductivity have provided both lower and higher individual results. This is due to the natural variability of the rock properties as well as to the fact that damage is correlated to the amount of explosives. The latter varies along the periphery of the opening and also along the longitudinal section of the tunnel. The properties of a damaged zone surrounding an underground opening change if spalling has occurred. The results of the major experiments suggest that the hydraulic conductivity will be in the order of 10^{-6} m/s and that such a zone extends a couple of decimetres into the rock surrounding the underground opening /Bäckblom 2009/.

Currently, there is no reliable direct method that can quantify the connected effective transmissivity along a tunnel, apart from judging the likelihood that no continuous EDZ has developed at all. SKB plans to develop several procedures for ensuring that the damage in deposition tunnels conforms to the design premises. Procedures to control and inspect the drilling, charging and ignition sequences will be developed and included in the monitoring and control programmes for the underground openings. The influence from rock conditions on the EDZ will be evaluated within the framework of the observational method and the associated monitoring programme, i.e. combining results from geological characterisation, geophysical techniques and geological modelling. Furthermore, the probability of spalling is reduced in the design as the deposition tunnels are aligned sub-parallel to the major horizontal stress.

Should the connected effective transmissivity caused by EDZ not conform to the design premises, the contingency action is to remove any loose rock debris that remains on the rock walls. The ultimate contingency measure is to reject the deposition tunnel.

In conclusion, there is ample evidence that the transmissivity of the EDZ will be kept below the accepted level stated in the design premises. Performed studies show that a continuous EDZ will not develop. In spite of this it is justifiable to assess how transmissive an EDZ needs to be in order to significantly impact other safety functions. The reason for this is that the occurrence of the EDZ currently can only be checked by indirect measurements. In addition, the local stress reduction close to the tunnel may open fractures parallel to it, as is further assessed in Section 10.2.2. Further discussion on these matters, together with input data for SR-Site, is found in the **Data report**, Section 6.5.

Deposition tunnel geometry

In Section 6.3.2 of the **Underground openings construction report**, it is concluded that the design premises on tunnel geometry, imposed by the backfill, are likely to be fulfilled using existing technology. However, deviations cannot be totally excluded and means of inspecting the geometry and handling the potential deviations are needed. Measuring techniques are well established and demonstrated to be fit for measuring the geometry of deposition tunnels after excavation. Should the as-built dimensions not conform to the design premises, the current contingency measure for maximum cross-section area and maximum volume is to apply shotcrete and smooth out the rock surface to cope with any irregularity or rock fall out. Should the design premises for minimum cross-section not be fulfilled due to occurrences of underbreak, the contingency measure is to remove it by using standard mechanical equipment. The ultimate contingency measure is to reject the deposition tunnel.

Grouting, reinforcement and stray materials

In Section 4.6 of the **Underground openings report** the following is concluded:

- According to the reference design, cement is used in shotcrete support, for embedding various rock support elements and in grout mixes for sealing purposes. Various recipes that would generate porewater with pH < 11 were developed for the purpose of grouting in rock types with potential for unacceptable inflows as well as for rock support elements that need to be embedded in cement. The composition and the amounts of the materials used for ground support and for grouting are given in Tables 4-1 and 4-2 of the **Underground openings construction report**.

- Quantities of stray material that remain in the rock from grouting activities will form part of the final repository facility. The assessed quantities on completion of the final repository facility are given in Table 4-3 of the **Underground openings construction report**. The estimated quantities contain cement, silica fume, silica sol and additives.
- The expected frequency of grouting is not foreseen to result in overlapping boreholes outside the tunnel perimeter to such an extent that they can be regarded as continuous relative to the length of the deposition tunnel.
- Well established and reliable, empirical, analytical and numerical design methodologies were used to assess the stability of underground openings and the rock support required for the underground openings in the reference design layout. The assessment of the deposition tunnels concludes that continuous shotcrete is not required for the foreseen rock qualities. The assessed composition and amounts of engineered materials associated with elements of rock support are presented in Table 4-1 of the **Underground openings construction report**.

Deposition tunnel inflows

In Section 6.3.2 of the **Underground openings repor**t it is concluded that the open flowing fracture frequency at the repository level, assessed to be around 0.005/m, indicates that, on average, less than 2% of the 20 m sections between deformation zones will require grouting /SKB 2009b/. However, there are uncertainties associated with the grouting methods and the level of effort to locate the water-bearing fractures, particularly if they have relatively small apertures with channelised flow. The estimated grout quantities required to reduce the inflows to acceptable levels are based on combining cement based grout mixes and solution grouting with silica sol. The assessed amounts of engineered material from grouting activities that would remain in the final repository are shown in Table 4-2 of the **Underground openings construction report**. Hence the confidence is high for detecting whether inflows to deposition tunnels conform to the design premises or not. If the inflow does not conform to the design premises post-grouting will be executed as a contingency measure. Details of the programme for **developing measurement techniques** for evaluating inflow of water to deposition tunnels are presented in /SKB 2010b/.

5.3 Initial state of the fuel and the canister cavity

5.3.1 Requirements on the handling of the spent nuclear fuel

All spent fuel from the currently approved Swedish nuclear programme shall be deposited in the KBS-3 repository. Only spent fuel in oxide form or with similar solubility is accepted for deposition in the KBS-3 repository. Furthermore, the **Spent fuel report** sets the maximum enrichment to 5% and the maximum average assembly burnup to 60 MWd/kgU for uranium oxide fuel (UOX) and 50 MWd/kgU for mixed oxide fuel (MOX) respectively.

There are a number of requirements on the handling of the spent fuel related to its final disposal in the KBS-3 repository, see the **Spent fuel report**, Chapter 3. The most important handling requirements and criteria related to long-term safety are the following.

- The fuel assemblies to be encapsulated in any single canister shall be selected with respect to burnup and age so that the total decay power in the canister will not result in temperatures exceeding the maximum allowed in the buffer. The total decay power in each canister must not exceed 1,700 W.
- The fuel assemblies to be encapsulated shall be selected with respect to enrichment, burnup, geometrical configuration and materials in the canister so that criticality will not occur during the handling and storage of canisters even if the canister is filled with water. The effective multiplication factor (k_{eff}) must not exceed 0.95 including uncertainties.
- Before the fuel assemblies are placed in the canister they shall be dried so that it can be justified that the allowed amount of water stated as a design premise for the canister is not exceeded. The amount of water left in any one canister shall be less than 600 g.
- Before the canister is finally sealed, the atmosphere in the insert shall be changed so that acceptable chemical conditions can be ensured. The atmosphere in canister insert shall consist of at least 90% argon.

- It shall be verified that the radiation dose rate on the canister surface will not exceed the level used as a premise in the assessment of the long-term safety. The radiation dose rate at the surface of the canister must not exceed 1 Gy/h.

5.3.2 Fuel types and amounts

BWR and PWR fuel

The major part of the nuclear fuel to be deposited consists of spent fuel from the operation of the twelve Swedish nuclear power plants, which are either of boiling water reactor (BWR) type or pressurised water reactor (PWR) type. The fuel types and amounts are derived from the spent fuel stored in Clab (31 December 2007) and a reference scenario for the future operation of the ten remaining power plants. In the reference scenario the operating times are set to 50 years for the four reactors at Ringhals and the three at Forsmark, and 60 years for the three reactors at Oskarshamn. The two reactors in Barsebäck were closed down after approximately 24 years and 28 years of operation, respectively. The majority of the fuel used in the reactors consists of uranium oxide fuel (UOX). From Oskarshamn, there will be minor amounts of mixed oxide fuel (MOX).

In order to estimate the number of spent fuel assemblies generated as a result of the future operation of the nuclear power plants the planned target burn-up of the fuel assemblies as well as the thermal power must be known. Generally the nuclear power plants plan for an increase in burnup from roughly 45 MWd/kgU to roughly 55 MWd/kgU.

The number of fuel assemblies discharged from the Swedish nuclear power plants stored in the interim storage facility at the end of 2007, and the estimated total number of assemblies for the reference scenario for future operation of the reactors are given in Table 5-2 together with the corresponding estimated amount of uranium or heavy metal expressed as tonnes initial weight. As seen in the table, the total amount of fuel generated in the reference scenario is around 11,100 tonnes (expressed as tonnes of U or heavy metal (HM) in initial fuel). It is noted that this is slightly less than the 12,000 tonnes (U or HM) which is the estimate used in the licence application.

Miscellaneous fuels

There are also minor quantities of other oxide fuel types from research and the early part of the nuclear power programme to be deposited in the KBS-3 repository. These fuels are in the following referred to as *miscellaneous fuels*.

The amount of miscellaneous fuel and, if applicable, the number of fuel assemblies in the interim storage facility as well as the estimated total amount of uranium or heavy metal, expressed as tonnes initial weight, is presented in Table 5-3.

Table 5-2. Spent fuel from operation of the nuclear power plants stored in the interim storage facility, Clab, and total amounts estimated for the SKB reference scenario. The information about fuel stored in Clab is based on Clab's safeguards accountancy system.

Fuel type	Number in interim storage (31 December 2007)	Total number for SKB reference scenario /SKBdoc 1221567/	Total initial weight for the reference scenario (tonnes of U or heavy metal HM)
BWR assemblies from operation of the NPP at Barsebäck, Oskarshamn, Forsmark and Ringhals	21,194[1,2]	47,498[3,4]	8,312[5]
PWR assemblies from operation of the NPP at Ringhals	2,552	6,016	2,791[6]

1) The fuel channels have been removed from 1,520 of the BWR assemblies.

2) Including 3 BWR MOX assemblies stored at Oskarshamn nuclear power plant.

3) Including 83 BWR MOX assemblies from Oskarshamn nuclear power plan.

4) Including rod cassettes, i.e. dismounted fuel rods placed in fuel rod cassettes.

5) Assumed 175 kg U/HM per assembly.

6) Assumed 464 kg U per assembly.

Table 5-3. Miscellaneous oxide fuels stored in the interim storage facility, Clab, and total amounts estimated for the SKB reference scenario. Information about fuel stored in Clab is based on Clab's safeguards accountancy system.

Fuel type	Number in interim storage 31 December 2007	Total number for the SKB reference scenario	Total initial weight for the reference scenario (tonnes of U or HM)
Fuel assemblies from the dismantled pressurised Ågesta heavy water reactor	222 (1 unirradiated)	222 (1 unirradiated)	20
Swap MOX assemblies (BWR)[1)]	184	184	14.1
Swap MOX assemblies (PWR)[1)]	33	33	8.4
Fuel residues in special boxes from Studsvik	19	Approximately 25	3
Damaged fuel in protection boxes[2)]	0	–	–

1) Some spent fuel from Ringhals and Barsebäck was earlier sent to La Hague for reprocessing. This spent fuel was exchanged in 1986 for spent German MOX fuel in equivalent amounts of plutonium.

2) Fuel assemblies damaged so that it is possible that material may fall off are emplaced in protection boxes during storage. Currently there is no such fuel. However, there are assemblies with leaking rods.

5.3.3 Handling

The handling of the spent fuel comprises the following main steps:

- Transport and deliveries of the fuel assemblies from the power plant to the interim storage facility, Clab (later Clink).
- Interim storage for typically 30–40 years.
- Selection of assemblies for encapsulation, transport to the encapsulation building and drying of the assemblies.
- Encapsulation consisting of placement in the canister and exchange of atmosphere in the canister insert prior to sealing of the copper canister.

The interim storage and encapsulation of the spent nuclear fuel will take place in Clink (Central interim storage and encapsulation plant). The Clink facility will consist of the current interim storage facility – Clab – and the encapsulation plant to be constructed in connection to Clab. Details of the different steps are given in the **Spent fuel report**.

5.3.4 Initial state

Encapsulated spent fuel is the spent nuclear fuel encapsulated for deposition in the KBS-3 repository. Gases and liquids in the cavities of the canister and fuel assemblies are considered as a part of the encapsulated spent fuel. The initial state of the encapsulated spent fuel refers to the properties of the spent fuel and the gases and liquids in the cavities of the canister when the canister is finally sealed and no further handling of the individual fuel assemblies is possible.

Radionuclide inventory

At closure of the final repository, the burn-up, irradiation and power history and age of the assemblies in each canister will be known and the radionuclide inventory can be calculated for each canister individually. However, at the present stage it is not possible to calculate the inventory in individual canisters. In order to give a reasonable account of the variability in inventory between canisters, a set of type-canisters has been defined subject to the condition that the total decay power of the encapsulated assemblies will not exceed 1,700 W. This condition will restrict the possible combinations of burnup and age of the assemblies and thus the variation in radionuclide inventory. The type-canisters provide a representative and adequate description of the canisters' content of fuel, its burn-up and age and the resulting radionuclide inventory in each canister.

The radionuclide inventory in a canister will depend on the number of assemblies in the canister, the burn-up of the assemblies and the age of the assemblies when they are encapsulated.

The burn-up and the number of assemblies will be the most important parameters for the radionuclide inventory. In a long-term perspective, the age of the fuel at deposition is of minor importance for the radionuclide inventory since the short lived nuclides of importance for the decay power will have decayed away.

The selected ages and burn-ups of the assemblies in the type canisters are based on the results of a simulation of the encapsulation procedure presented in Section 5.2 of the **Spent fuel report**. The following type canisters have been selected as a basis for descriptions of the radionuclide inventory in individual canisters:

- BWR I: representing the largest part of the BWR canisters.
- BWR II: representing the high end radionuclide inventory in BWR canisters.
- BWR III: representing the partially filled BWR canisters[9].
- BWR-MOX: representing BWR canisters containing MOX assemblies.
- PWR I: representing the largest part of the PWR canisters.
- PWR II: representing the high end radionuclide inventory in PWR canisters.
- PWR III: representing the partially filled PWR canisters.
- PWR-MOX: representing PWR canisters containing MOX assemblies.

The total numbers of the different type canisters is illustrated in Figure 5-7.

To verify that the type canisters provide a representative and adequate description of the total radionuclide inventory in the repository, the summed total inventory in all type canisters has been compared with the total radionuclide inventory calculated for the reference scenario for the operation of the nuclear power plants. The resulting total inventories of thirteen radionuclides of importance for decay power, radiotoxicity and calculated long-term risk and the summed inventory of all nuclides are presented in Table 5-4. The table includes the summed inventory in all the different type canisters as well as in all canisters in the repository independent of type. As seen in the table, the total inventory in the type canisters slightly exceeds that of the reference scenario, justifying this aspect of the type canister approach. More detailed results of this calculation are given in the **Spent fuel report**.

Table 5-4. The total inventories (Bq) of thirteen radionuclides of importance for decay power, radiotoxicity and calculated long-term risk in all type canisters and all canisters in the repository (calculated for the year of encapsulation) compared to the total inventory calculated for the calendar year 2045 for the reference scenario for the operation of the nuclear power plants (Table 6-13 in the Spent fuel report). Note that the calculations in SR-Site include many more nuclides than those listed in this table. The full inventory is provided in the Data report.

Radionuclide	BWR I	BWR II	BWR III	BWR-MOX	PWR I	PWR II	PWR III	PWR-MOX	Total in all type canisters	Total for the reference scenario
Am-241	$6.6\cdot10^{17}$	$1.1\cdot10^{17}$	$3.9\cdot10^{17}$	$1.2\cdot10^{17}$	$3.3\cdot10^{17}$	$1.5\cdot10^{16}$	$1.5\cdot10^{17}$	$1.2\cdot10^{16}$	$1.8\cdot10^{18}$	$1.7\cdot10^{18}$
C-14	$2.2\cdot10^{14}$	$3.7\cdot10^{13}$	$1.5\cdot10^{14}$	$2.6\cdot10^{13}$	$7.3\cdot10^{13}$	$3.3\cdot10^{12}$	$3.6\cdot10^{13}$	$2.2\cdot10^{12}$	$5.5\cdot10^{14}$	$5.2\cdot10^{14}$
Cl-36	$1.0\cdot10^{12}$	$1.9\cdot10^{11}$	$7.3\cdot10^{11}$	$1.1\cdot10^{11}$	$3.4\cdot10^{11}$	$1.7\cdot10^{10}$	$1.8\cdot10^{11}$	$9.6\cdot10^{9}$	$2.6\cdot10^{12}$	$2.3\cdot10^{12}$
Cs-137	$9.3\cdot10^{18}$	$1.2\cdot10^{18}$	$6.9\cdot10^{18}$	$9.3\cdot10^{17}$	$4.2\cdot10^{18}$	$1.3\cdot10^{17}$	$2.1\cdot10^{18}$	$1.3\cdot10^{17}$	$2.5\cdot10^{19}$	$2.3\cdot10^{19}$
I-129	$5.6\cdot10^{12}$	$9.8\cdot10^{11}$	$3.8\cdot10^{12}$	$6.7\cdot10^{11}$	$2.6\cdot10^{12}$	$1.2\cdot10^{11}$	$1.4\cdot10^{12}$	$7.5\cdot10^{10}$	$1.5\cdot10^{13}$	$1.4\cdot10^{13}$
Nb-94	$2.0\cdot10^{13}$	$3.4\cdot10^{12}$	$1.3\cdot10^{13}$	$2.3\cdot10^{12}$	$6.1\cdot10^{14}$	$2.6\cdot10^{13}$	$2.8\cdot10^{14}$	$1.9\cdot10^{13}$	$9.8\cdot10^{14}$	$9.3\cdot10^{14}$
Pu-238	$5.3\cdot10^{17}$	$1.0\cdot10^{17}$	$4.4\cdot10^{17}$	$6.9\cdot10^{16}$	$2.5\cdot10^{17}$	$1.3\cdot10^{16}$	$1.7\cdot10^{17}$	$8.0\cdot10^{15}$	$1.6\cdot10^{18}$	$1.3\cdot10^{18}$
Pu-239	$5.6\cdot10^{16}$	$8.0\cdot10^{15}$	$3.1\cdot10^{16}$	$8.8\cdot10^{15}$	$2.6\cdot10^{16}$	$9.6\cdot10^{14}$	$1.1\cdot10^{16}$	$8.5\cdot10^{14}$	$1.4\cdot10^{17}$	$1.4\cdot10^{17}$
Pu-240	$1.0\cdot10^{17}$	$1.7\cdot10^{16}$	$6.6\cdot10^{16}$	$1.7\cdot10^{16}$	$4.2\cdot10^{16}$	$1.9\cdot10^{15}$	$2.1\cdot10^{16}$	$1.6\cdot10^{15}$	$2.7\cdot10^{17}$	$2.5\cdot10^{17}$
Pu-241	$4.0\cdot10^{18}$	$3.7\cdot10^{17}$	$3.2\cdot10^{18}$	$4.5\cdot10^{17}$	$1.9\cdot10^{18}$	$3.5\cdot10^{16}$	$1.0\cdot10^{18}$	$7.2\cdot10^{16}$	$1.1\cdot10^{19}$	$1.1\cdot10^{19}$
Sr-90	$6.2\cdot10^{18}$	$7.6\cdot10^{17}$	$4.4\cdot10^{18}$	$6.0\cdot10^{17}$	$2.8\cdot10^{18}$	$8.0\cdot10^{16}$	$1.3\cdot10^{18}$	$8.2\cdot10^{16}$	$1.6\cdot10^{19}$	$1.6\cdot10^{19}$
U-234	$2.2\cdot10^{14}$	$3.7\cdot10^{13}$	$1.2\cdot10^{14}$	$3.0\cdot10^{13}$	$1.3\cdot10^{14}$	$5.5\cdot10^{12}$	$5.4\cdot10^{13}$	$3.4\cdot10^{12}$	$6.1\cdot10^{14}$	$6.0\cdot10^{14}$
U-238	$5.4\cdot10^{13}$	$7.8\cdot10^{12}$	$3.0\cdot10^{13}$	$6.5\cdot10^{12}$	$2.2\cdot10^{13}$	$8.0\cdot10^{11}$	$8.8\cdot10^{12}$	$6.3\cdot10^{11}$	$1.3\cdot10^{14}$	$1.3\cdot10^{14}$

[9] In order to keep the maximum thermal output below 1,700 W, canisters containing high decay power fuel elements cannot utilise their full space.

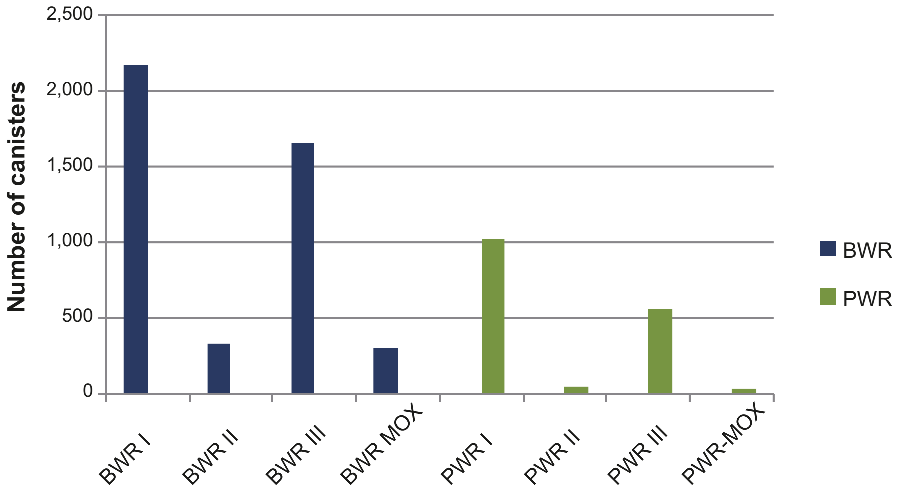


Figure 5-7. *The total numbers of the different kinds of type canisters.*

The radionuclide inventory in the different type canisters of some of the more important radionuclides are presented in Table 5-4. Full inventories are given in the **Spent fuel report** and in the **Data report**. The radionuclides in the inventory are found in the fuel, construction materials in the fuel assembly, crud (a deposit on the cladding), and control rods (PWR canisters).

A fraction of the inventory is located in parts which are assumed to rapidly release the radionuclides in the event of a canister failure. This fraction includes radionuclides located in all parts except the fuel matrix. Certain elements, originally formed in the fuel matrix, have high enough mobility that a fraction of the inventory will transfer to the fuel/cladding gap and thereby contribute to the IRF. The size of this fraction can, for some elements, be estimated from measured and calculated fission gas (Xe, Kr) release (FGR). The FGR is strongly correlated to the linear heat generation rate, which in turn will depend on the thermal effect of the nuclear reactor, the number of assemblies and configuration of the assemblies in the reactor core and on how the fuel assemblies are utilized during operation. The fission gas release of the assemblies in the different type canisters has been estimated based on the average burn-up of the assemblies in the canisters. The resulting estimated FGR for the assemblies in the type canisters is given in Table 5-5.

Of all relevant radionuclides Cl-36 is among the most mobile and the fraction of Cl-36 in the fuel/cladding gap is about three times the FGR. Most significantly, the fraction of the mobile radionuclides I-129, Cs-135, and Cs-137 is found to have an IRF which is, at most, equal to the FGR. Other, less mobile radionuclides originally formed in the fuel matrix (e.g. Se-79, Sr-90, Tc-99, Pd-107, Sn-126) are also found to have an IRF; however, available leaching data indicate that this fraction is very small and not correlated to FGR. The actual IRF values used in the safety assessment modelling for all nuclides are given in the **Data report**.

Table 5-5. Estimated FGR of the assemblies in the different type canisters.

BWR canisters		PWR canisters	
Type canister	FGR (%)	Type canister	FGR (%)
BWR I	1.3	PWR I	2.9
BWR II	2.1	PWR II	7.8
BWR III	2.1	PWR III	7.8
BWR-MOX	2.3	PWR-MOX	2.9

Criticality

The assemblies must not under any circumstances be encapsulated if the criticality criteria cannot be met. Loading curves, describing burn-up in relation to enrichment to stay sub-critical, for checking for criticality have been calculated for both BWR and PWR canisters. When calculating the loading curves, all uncertainties regarding the propensity for criticality have been systematically investigated according to principles generally applied for all handling of nuclear fuel /SKBdoc 1193244/. Before encapsulation, each individual assembly is checked against the loading curve. If an individual assembly does not conform to the acceptance criteria in this check, an inspection of the specific set of assemblies selected for encapsulation will be made. If it cannot be shown that the selected combination of assemblies conforms to the criticality criteria, a new selection will be made. Should it not be possible to combine an individual assembly with other assemblies to conform to the criticality criterion, it can be encapsulated alone in a canister. The ultimate measure will be to alter the geometry, i.e. to reconstruct the assembly.

The criticality calculations are based on the canister reference design, which is constrained by design premises related to criticality and expressed as requirements on geometry and material composition, see Section 5.4.

Variables for the spent fuel

In the **Fuel and canister process report** the spent fuel is described by the variables in Table 5-6, which together characterise the spent fuel in a suitable manner for the safety assessment. The description applies not only to the spent fuel itself, but also to the cavities in the canister. This section presents the initial condition values for the variables.

Geometry

The geometries of representative BWR and PWR assemblies are presented in the **Spent fuel report**. The BWR fuel assemblies contain about 60 up to 100 fuel rods. The fuel rods consist of zirconium alloy tubes filled with cylindrical fuel pellets. The rods are arranged in square arrays enclosed in a fuel channel. The cross-sectional area of the fuel assemblies is about $0.141 \cdot 0.141\ m^2$ and the total length can be up to about 4.4 m. The PWR fuel assemblies contain 204 or 264 fuel rods, arranged in square arrays. The cross-sectional area is about $0.214 \cdot 0.214\ m^2$ and the total length is about 4.3 m.

Table 5-6. Variables for fuel/cavity in canister.

Variable	Definition
Geometry	Geometric dimensions of all components of the fuel assembly, such as fuel pellets and Zircaloy cladding. Also includes the detailed geometry, including cracking, of the fuel pellets.
Radiation intensity	Intensity of alpha, beta, gamma and neutron radiation as a function of time and space in the fuel assembly.
Temperature	Temperature as a function of time and space in the fuel assembly.
Hydrovariables (pressure and flow)	Flows, volumes and pressures of water and gas as a function of time and space in the cavities in the fuel and canister.
Mechanical stresses	Mechanical stresses as a function of time and space in the fuel assembly.
Radionuclide inventory	Occurrence of radionuclides as a function of time and space in the different parts of the fuel assembly. The distribution of the radionuclides between the pellet matrix and surface is also described here.
Material composition	The materials of which the different components in the fuel assembly are composed, excluding radionuclides.
Water composition	Composition of water (including any radionuclides and dissolved gases) in the fuel and canister cavities.
Gas composition	Composition of gas (including any radionuclides) in the fuel and canister cavities.

Radiation intensity

The highest obtained radiation dose rate stated in the **Spent fuel report** is 0.18 Gy/h on limited areas of the canister. This level was calculated for PWR canisters containing four assemblies with a burn-up of 60 MWd/kgU and an age of 30 years, which would give a heat output much higher than will be allowed in the repository. For a PWR element with a burn-up of 42.2 MWd/kgU after 34.1 years, according to the specifications for the type canisters, the source strength of neutrons is $1.2 \cdot 10^8$/s and of photons $6.8 \cdot 10^{14}$/s. A canister with four such elements (1,700 watt) will have a maximum dose rate on the canister surface of 115 mSv/h and a maximum neutron dose rate of 2.9 mSv/h.

Temperature, Hydrovariables (pressure and flow) and Mechanical stresses

The initial conditions of these variables are dependent on the external environment and are assessed where appropriate in SR-Site. Only the temperature is of relevance for an intact canister.

Material composition

An overview of the materials in typical BWR and PWR fuel assemblies is given in Table 5-7. More detailed information is provided in the **Spent fuel report** and in the **Data report**.

Four elements of particular interest for the assessment of long-term safety are nitrogen (N), chlorine (Cl), nickel (Ni) and niobium (Nb) since these, through neutron capture during operation, form radioactive isotopes. The contents of these elements in the construction materials for a PWR Areva 17·17 assembly and a BWR Svea 96 Optima 2 assembly are given in Table 5-8. The variability of these elements in the different kinds of BWR and PWR assemblies to be deposited has been investigated by randomly selecting a number of fuel types and comparing the amounts of construction materials in these assemblies with the amounts in Svea 96 Optima 2 and Areva 17·17 respectively. The conclusion is that the amounts will be similar for all BWR and PWR elements, respectively.

Water and gas composition

The content of gases and liquids depends on the result of the drying and gas exchange. There is no reason to believe that the water content in the canisters will exceed the 600 g given as a premise or that the argon content will be below the acceptable level of at least 90%.

Table 5-7. Overview of the materials in typical BWR and PWR fuel assemblies (Data based on specifications provided by the nuclear fuel suppliers).

Material composition	BWR Svea 96 Optima 2	PWR Areva 17·17
	Weight in 1 fuel assembly (kg)	
Fuel		
U-tot	175	464
O	23	62
Cladding material		
Zirconium alloys	49	108
Stainless steel	–	3
Fuel channel		
Zirconium alloys	32	–
Stainless steel	8	–
Other constructions (bottom and top plate, spacers etc)		
Stainless steel	5	12
Zirconium alloys	–	21
Nickel alloys	1	2

Table 5-8. The content of N, Cl, Ni and Nb in the construction material for typical BWR and PWR fuel assemblies.

Element	Weight in 1 fuel assembly (kg)						
	BWR Svea 96 Optima 2				PWR Areva 17·17		
	Cladding	Other	Fuel channel	Total[1]	Cladding	Other	Total[1]
N	0.002	0.002	0.005	0.009	0.0055	0.0058	0.011
Cl		$6 \cdot 10^{-6}$	$8 \cdot 10^{-6}$	$1.4 \cdot 10^{-5}$	$3 \cdot 10^{-6}$	$1.4 \cdot 10^{-5}$	$1.7 \cdot 10^{-5}$
Ni	0.02	1.1	0.86	1.99	0.27	2.19	2.46
Nb		0.0082	$8 \cdot 10^{-4}$	0.0091	1.08	0.09	1.17

1) Total sometimes appears to differ from the sum of the individual contributions due to rounding.

5.4 Initial state of the canister

5.4.1 Design premises relating to long-term safety

The process of defining the design premises for the canister is based on the overall strategy discussed in Section 5.1.1 and as described in Chapter 2 of the **Canister production report**. In the following the main design premises on the canister, as defined in the Design premises report /SKB 2009a/, are listed.

- "The canister shall withstand an isostatic load of 45 MPa, being the sum of maximum swelling pressure and maximum groundwater pressure.
- The copper corrosion barrier should remain intact after a 5 cm shear movement at a velocity of 1 m/s for buffer material properties of a 2,050 kg/m^3 Ca-bentonite[10], for all locations and angles of the shearing fracture in the deposition hole, and for temperatures down to 0°C. The insert should maintain its pressure-bearing properties to isostatic loads.
- A nominal copper thickness of 5 cm, also considering the welds.
- The spent fuel properties and geometrical arrangement in the canister should further be such that criticality is avoided if water should enter a canister."

The design premise for the canister to withstand isostatic load also encompasses asymmetric loads due to uneven swelling of the bentonite. Temporary asymmetric loads could occur due to uneven saturation of the buffer or due to irregularities in the deposition hole. Irregularities in the deposition hole may also result in permanent asymmetric loads.

There are also some more indirect requirements on material composition and structure, and on the environment in the sealed canister. These are summarised as follows.

- "Gamma radiation causes hardness and brittleness in cast iron. Copper content in cast iron < 0.05%.
- Creep ductility: Grain size < 800 µm, Phosphorus content in copper: 30–100 ppm, Sulphur content in copper < 12 ppm.
- Brittleness of copper: Hydrogen content < 0.6 ppm.
- In addition: oxygen content of some tens of ppm can be allowed in the copper. However the material used for trial production has had a requirement < 5 ppm and before a change can be made further testing is needed.
- The quantity of nitric acid that can be formed in the insert shall be limited by replacing the atmosphere in the insert by > 90% argon. The permissible water quantity in the insert is set to 600 g."

There are also design premises on the canister from the spent fuel, which is a further specification of the requirement to prevent criticality:

- The material composition of the nodular cast iron shall be: Fe > 90%, C < 4.5% and Si < 6%.

The production and operation gives the following design premises for the copper canister:

- To allow ultrasonic testing the average grain size shall be less than 360 µm.

[10] It should be noted that the design premise concerns a Ca-bentonite since it is pessimistically assumed that the reference bentonite material will be transformed to the stiffer Ca-bentonite before the shearing event.

5.4.2 Reference design and production procedures

The reference design specifies the current design of the canister. The reference design shall conform to the design premises and be demonstrated as technically feasible to produce by the methods for production and the methods applied for encapsulation, transportation and deposition. The reference design is given in detail in Chapter 3 of the **Canister production report** and the production and inspection procedures are described in Chapter 5 in the **Canister production report**. A short summary is provided below in order to demonstrate that the production and inspection process leads to a canister within the limits provided by the reference design parameters, such that it conforms to the design premises.

Reference design

The reference design is described by a set of design parameters for which nominal values and acceptable variations are given. In short, the current reference design of the canister consists of a tight corrosion barrier of copper and a load-bearing insert of nodular cast iron. The sealed canister has a total length of 4,835 mm and a diameter of 1,050 mm, see Figure 5-8. The copper shell thicknesses with acceptable tolerance for the reference design are:

- Copper tube: 49±0.3 mm.
- Welds: 48.5±0.7 mm.
- Lid: 50±0.6 mm.
- Base: 50±1.0 mm.

Two reference canister designs have been developed, one for 12 BWR fuel assemblies and one for 4 PWR fuel assemblies, Figure 5-8. The maximum total weight of the canister, including fuel, is 24,600 kg for BWR and 27,000 kg for PWR. Other types of fuel assemblies, see Section 5.3.2, for example swap MOX assemblies and fuel assemblies from Ågesta, can be accommodated in these canister designs.

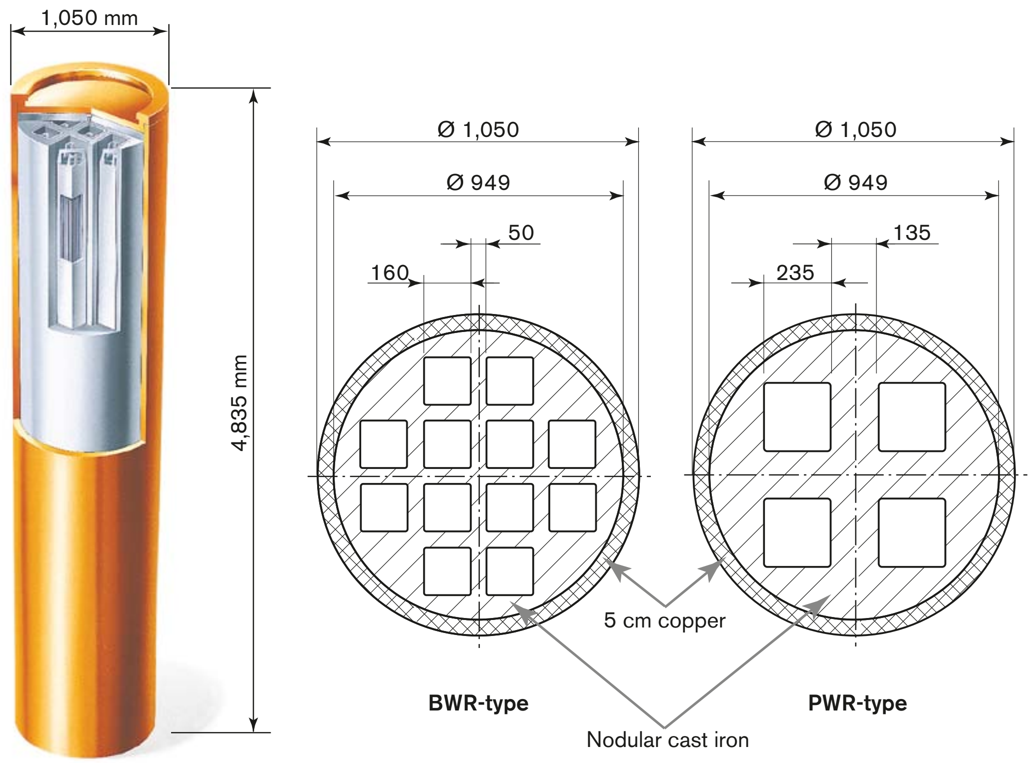


Figure 5-8. *Left: The reference design with a corrosion resistant outer copper shell and a load-bearing insert of nodular cast iron. Right: Cross section of insert designs of the BWR and PWR types.*

The canister comprises the cast iron insert with a steel tube cassette for each fuel element, the steel lid, the copper tube, the copper lid and the copper base, see Figure 5-9.

The verifying analyses discussed in Chapter 4 of the **Canister production report**, have been used to identify which of the canister properties are important for the different load cases and to specify values of the related design parameters for which the reference design conforms to the design premises.

In addition, for both the isostatic and the shear load cases, damage tolerance analyses are carried out on the reference design to establish acceptable sizes of defects in the materials. It must then be demonstrated that the production and inspection procedures yield canisters conforming to these specifications.

For the isostatic load, the damage tolerance analyses show that defects about 20 mm large are acceptable in the outer part of the insert whereas larger defects are acceptable in the interior part along the entire length, i.e. such defects do not grow in a way that jeopardises the integrity of the inserts for an isostatic load of 45 MPa. For the bottom part of the insert, axial crack-like defects with a depth of up to 80% of the material thickness can be allowed.

For the shear load, the maximum accepted depth for crack-like surface defects in the circumferential direction is 4.5 mm for semi-elliptical shape and 8.2 mm for semi-circular shape. The insert is less sensitive to internal defects, allowing defects larger than 10 mm. The analyses also show that the results are clearly dependent on the buffer density. A lower buffer density means that larger defects can be accepted in the insert.

Production and inspection of the inserts and steel lid

The inserts are cast with 12 channels for BWR assemblies or 4 channels for PWR assemblies. The eccentricity of the cassette in the insert is inspected by measuring the position of the cassette, in relation to the insert envelop surface, at both the top and the bottom of the insert. Exterior machining and ultrasonic testing are performed in steps to inspect the inserts, followed by a final surface inspection.

A key dimension for the mechanical strength needed to withstand the maximum isostatic load is the edge distance of the insert (i.e. the distance between the external corners of the channel tubes and the insert casing surface), see Figure 5-8. The specified edge distance is 33.3 ±10 mm. The results from the test manufacturing shows that manufactured inserts conform to the specification (misalignment of 3–8 mm). The misalignment under normal production is assumed to be ±5 mm. The probability of exceeding the specified ±10 mm is regarded as negligible based on the fact that the eccentricity of the cassette is inspected by ultrasonic testing after casting, and again after machining if the cassette was not centered.

Regarding defect sizes, results from the test manufacturing of inserts show that the material is homogeneous. In the inserts in the demonstration series, only clustered porosity has been found (no large defects detected) and only located in areas where rather large defects (~40 millimetres along the whole length) are acceptable. The non-destructive testing methods have 90% probability of detecting volumetric defects of c. 5 mm for the near-surface region (down to 50 mm) and below 10 mm for the deeper regions. (These detection capabilities have been determined with a 95% confidence range.) For the isostatic load these margins of the non-destructive testing (NDT) methods for detecting the acceptable defect size imply that the probability for missing critical defects during inspection can be considered low.

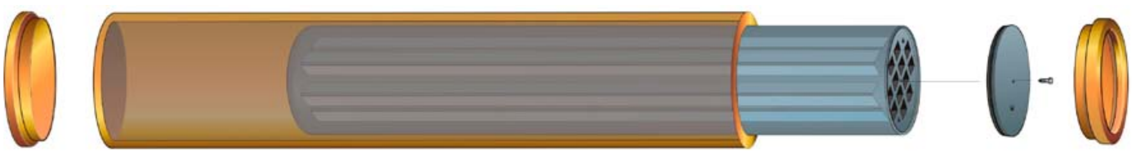


Figure 5-9. Exploded view of the canister components (from left: copper base, copper tube, insert, steel lid for insert and copper lid).

In the analyses of the BWR demonstration series no crack-like defects were detected. The probability of detection for the NDT, using the current preliminary inspection procedures, is, for artificial crack-like defects, 2–3 mm for surface defects and 4–9 mm for internal defects. However, the occurrence of real defects oriented other than in radial-circumferential directions cannot be stated as no adapted inspection techniques have been developed. For the shear load case the damage tolerance analysis gives acceptable defect sizes, see Section 5.4.3, that put rigorous requirements on manufacturing and NDT capability for the insert. Based on the results and experience so far it is expected that these additional requirements can be implemented in production and the testing methods for verification.

The steel lids will be manufactured from steel plates and delivered with certificates that verify that the yield tensile strength conforms to the design parameter specifications. This parameter has not been further tested and inspected in the test production.

Production and inspection of the copper components

Extrusion is the reference method for the hot forming of the copper tubes. At the canister factory the extruded tube is then ultrasonically tested, and finally machined to the dimensions specified in the drawings. A concluding inspection after final machining with some form of surface inspection will be carried out. The lids and bases of copper are manufactured by forging, yielding a blank that is machined into a lid or a base. The machining of blanks into lids and bases is performed in two stages with intermediate non-destructive testing.

Machining is a well-known and proven industrial process. Therefore, the methods for inspection of copper thickness have not been determined and investigated in detail by SKB. Generally, mechanical inspection systems have measurement uncertainties less than 0.1 mm. One option is to use ultrasonic testing and at present, the measurement uncertainties are estimated to be 0.3 mm for 50 mm thick copper. Based on these data, the minimum copper thickness after machining in normal operations is judged to be 48.4 mm for the tube, 48.7 for the base, 48.1 for the lid, including a 1 mm deep identity marking, and 47.5 mm for the welds, i.e. the overall minimum thickness in normal operation is 47.5 mm, see Table 5-9.

Furthermore, it can, at this stage of development of the canister production, not be excluded that a few canisters per thousand might have areas where the minimal copper thickness is reduced to 45 mm due to disturbed operation, Table 5-9. This value is based on the fact that larger deviations will be detected when the components are handled, assembled and the canister is sealed and these deviations will thereby lead to rejection of the canister. Experience indicates that such large deviations in dimensions can even be detected with the naked eye. The probability that any canister will have a minimum thickness after machining of less than 45 mm is therefore judged as negligible.

Welding and inspection of the welding

The lids and the bases are welded with the same method, friction stir welding (FSW). Welding of the bases is carried out at the canister factory whereas welding of the lids (sealing) is performed at the encapsulation plant.

Control of a set of process parameters in connection to the welding ensures that it has been conducted within a so-called process window, which has been shown to result in welds according to specification. The welding process has relatively few process parameters, and it has been proved possible to both control and monitor them. The process is normally very stable with a high level of repeatability since the process is adaptive, i.e. the important tool temperature is measured constantly and the input parameters are adjusted in order to keep the temperature within a given range.

The occurrences of defects in the welds have been investigated based on the results of two welding demonstration series under normal operational conditions performed at the Canister Laboratory. Normal operation during welding means that both input and output welding variables are within a defined "process window" that must be met in order for a weld to be considered approved. The only type of defect that has been detected in the welds made in normal operation is joint line hooking (a tight crack like discontinuity with extension radially with a gap of < 10µm, see Figure 5-10). The wormhole type of defect (near-surface discontinuity) has only been seen outside normal operations for the weld.

Table 5-9. Minimum thicknesses of the copper shell after machining, i.e. a copper shell without defects is at least this thick over its entire surface. Defects can locally reduce the thickness. Further information is given in Sections 5.3.9, 5.4.9, 5.5.7 and 7.1.5 of the Canister production report.

Design parameter	Reference design (mm)	Initial state value (all dimensions in mm)
Thickness (mm) – tube – lid, base – welds	 49.0 50.0 48.5	 Minimum > 47.5 Fraction of canisters > 99% 45–47.5* Fraction of canisters: few per thousand Minimum < 45* Fraction of canisters: negligible
Local reduction in thickness due to defects (mm)	–	< 10 Fraction of canisters: > 99.9% 10–20* Fraction of canisters: one per thousand > 20* Fraction of canisters: negligible

*Values occurring only in disturbed operations considering both the manufacturing processes and inspection.

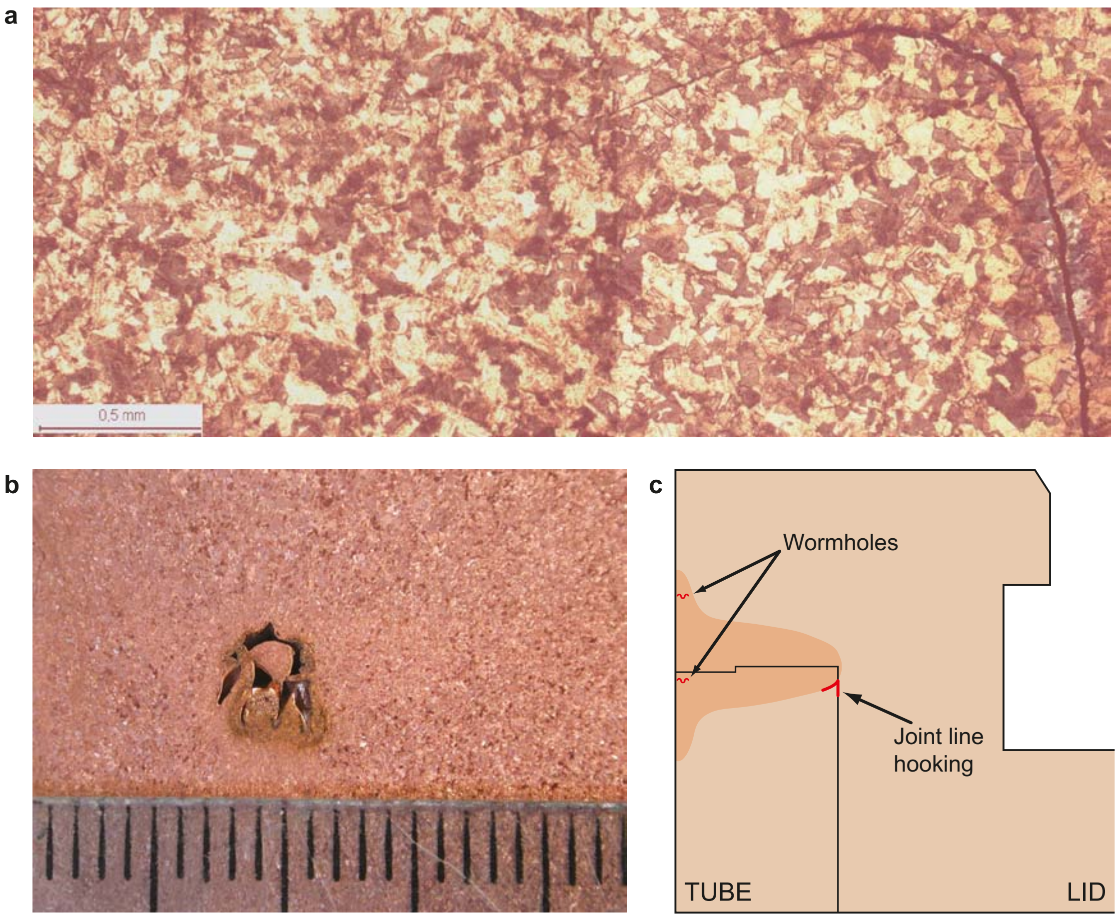


__Figure 5-10.__ a) Photograph of joint line hooking (from /SKBdoc 1175162/). b) Photograph of wormhole (from /SKBdoc 1175162/). c) Schematic cartoon showing positions of defects in friction stir welds (redrawn from the __Canister production report__).

In the first demonstration series, the largest detected defects were of the order of a few millimetres with the largest one being 5.4 mm in radial direction, i.e. in the direction of the corrosion barrier. The second series was performed with an improved welding tool and resulted in smaller defects with the largest one being 1.5 mm.

The evaluation of the reliability of the welding process, of its surveillance functions and of the NDT resulted in a maximum copper thickness reduction in the welds, during normal operations, of 10 mm for a population of 12,000 welds, corresponding to 6,000 canisters, Table 5-9. The results from probability-of-detection studies of the NDT methods show a detection capability (with 95% confidence limit), for the joint line hooking, of 90% for a 4 mm defect and close to 100% for a 10 mm defect.

The probability for disturbed operations, i.e. that one or more of the process variables are outside the process window is estimated to be low. The probability that the process variables are such that they cause defects that exceed 10 mm and that this occurs simultaneously with a failure of the control system, is at the present stage of development judged to be below one per thousand canisters. This statement is based on the fact that the developed welding process is very reliable and reproducible. If these disturbed operations do occur, the maximum reduction of the copper thickness is estimated not to exceed 20 mm, being the sum of a maximum joint line-hooking defect of 10 mm in the radial direction extending over a considerable part of the canister circumference in combination with wormholes (that should be clearly visible) with the same size. See Table 5-9.

For the rest of the copper shell, i.e. the base, the lid and the tube, available data on observed defects is limited. All copper components manufactured between January 2007 and May 2008, eight tubes and twenty lids, have been inspected by the preliminary NDT methods developed at the Canister Laboratory. The results of these inspections indicate that there are no defects in the extruded tubes or forged lids and bases that remain after the final machining.

Since the possible defects in the copper components are expected to mainly extend perpendicularly to the corrosion barrier and since no real manufacturing defects have been found in the canister components, the welds are presently considered to be the potentially thinnest parts of the copper shell.

Reductions may also occur as a result of surface damage during transportation, handling and deposition of the canister. Available information on the occurrence of surface damage during these stages is limited since full-scale tests, which focus on this issue using relevant handling equipment and transport casks, remain to be performed. The probability for critical reduction of the corrosion barrier due to transportation damages is, however, considered negligible since the canister is protected by a transport cask. In addition, the canister is inspected for surface damage when it is lifted from the transport cask into the radiation shield of the deposition machine in the reloading station at repository depth.

The material composition is determined by the manufacturing of the copper ingots together with the following hot-forming and welding processes. The copper quality used in the ingots is a pure oxygen-free copper of a standard quality. The material composition in the welds is determined primarily from the basic material but the welding process may be influenced by, e.g. tracers being carried over from the welding tool and changes in the oxygen content. The machining of the components and welds has no influence on the material composition.

Analyses of the copper ingot performed during the test manufacturing shows that the oxygen content in the ingots is well below the technical specification of maximum 5 ppm.

The material composition in the welds has been examined in six samples taken from three welds from the welding demonstration series and in three samples from one weld produced in argon gas. Analyses of the welds produced in air showed a mean oxygen content (with standard deviation) of 11.1 ± 12.2 ppm. The high standard deviation can be explained by the highest measured oxygen content that varies from 9 ppm in the steady-state sequence to 44 ppm in the overlap sequence. In the weld produced in argon gas the highest measured oxygen content was 2 ppm and the mean oxygen content (with standard deviation) was 1.8 ± 0.4 ppm.

To summarise, the values from the weld produced in argon gas are well below some tens of ppm stated in the design premises, whereas the welds produced in air have increased oxygen content in the overlap sequence. It should be noted that the welding procedure is currently being developed to minimise oxygen content. In addition, the influence of oxides formed at the inner lid/tube interface (at the root of the weld zone) on weld integrity is under investigation.

In addition to oxygen, the welds also include traces of tool material. A clear reduction of this contaminant has been seen by introduction of a tool with PVD (physical vapour deposition) coating of chromenitride (CrN). The levels of Ni, Co and Cr are all less than 1 ppm to be compared to the mean levels (with standard deviation) when using non coated tools (18 samples from 3 welds) of Ni, Co and Cr that were 6.0 ± 6.8 ppm, 1.8 ± 2.8 ppm and 1.6 ± 1.8 ppm respectively.

A corrosion study /Gubner and Andersson 2007/ on the weld zones from welds made in air and with non-coated tools was performed and concluded that the FSW tool is cathodic compared to the copper – small particles in the weld are cathodic protected by surrounding copper, resulting in a very small cathode compared to large copper anode. The good corrosion resistance of the FSW tool material will even further reduce the risk of corrosion of the surrounding weld material. Therefore, small metallic particles from the FSW tool do not pose a risk for accelerated corrosion of the welds. The study also concluded that a negative effect of copper oxides close to the surface could not be detected.

Canister assembly

The final stage of canister manufacture consists of the insertion of the insert into the copper tube complete with base. Assembly of the canister is performed in the canister factory.

Encapsulation and inspection of encapsulated canisters

The encapsulation of spent nuclear fuel is performed in the encapsulation plant. The nuclear fuel assemblies are selected and dried before being placed in the canister. The insert steel lid is mounted and the atmosphere in the insert is replaced with argon. Thereafter the canister is sealed by welding of the copper lid. The welds are inspected before the canister leaves the encapsulation plant. The canister is placed in a transport cask and transported to the final repository, where it is transferred to a deposition machine and placed in the deposition hole. The previous section summarises SKB's experiences with the welding.

5.4.3 Initial state

The initial state refers to the properties of the canisters once they have been deposited in the final repository and will not be further handled within the repository facility, see Section 5.1. The conclusions regarding the initial state, in particular conclusions regarding conformity to the design premises are derived in several steps in the **Canister production report**.

In a first step, the conformity of the reference design to the design premises is evaluated through a number of analyses compiled in /Raiko et al. 2010/ and summarised in Section 4.11 in the **Canister production report**. In a second step the capability of the canister production line to produce canisters that conform to the reference design is evaluated. The canister production line consists of manufacturing of the canisters, including the assembly of canisters in the canister factory, and of the encapsulation, transportation and deposition. These activities are summarised in Chapters 5 and 6 in the **Canister production report**, respectively.

Finally, the conformity of the deposited canisters to the reference design is evaluated and then the conformity of the canister at the initial state to the design premises is concluded in Sections 7.1 and 7.2 respectively in the **Canister production report**, essentially by referring to the results of the previous steps.

Isostatic load in the repository

The design premise states that the canister shall withstand an isostatic load of 45 MPa, being the sum of maximum buffer swelling pressure and maximum groundwater pressure.

It is stated in the **Canister production report**, Section 7.2.1, that the probability that the canister would not fulfil the design premises related to isostatic loads is insignificant. The basis for this judgement can be summarised as follows.

- The collapse load for the cylindrical part of a canister (without defects) has been calculated to be 99 MPa and 128 MPa for BWR and PWR inserts, respectively /Raiko et al. 2010/. Results from pressure tests of real canisters show that the collapse load for the canister is approximately 100 MPa or higher /Raiko et al. 2010/.

- An analysis of the reference design of the canister where the variability of the material data (yield stress, ultimate strength and fracture toughness) was treated probabilistically shows that the likelihood for local plastic collapse or initiation of crack growth due to variation in material properties is insignificant. The probability of local plastic collapse or initiation of crack growth is below $1 \cdot 10^{-50}$ (Section 4.3.1 in the **Canister production report**).
- The allowed defect sizes based on the damage tolerance analysis are large and the full scale test manufacturing trials show that inserts can be produced that have large margins with respect to defect sizes. In addition, the detection capabilities of the non-destructive testing methods are sufficient to identify the allowed defects with margins.
- The material properties obtained in the test manufactured components are homogeneous and conform to the values specified for the design parameters of the reference design, and the inspection of the material properties is performed by well-proven techniques.
- The edge distance of the cassette within the insert, which is a critical dimension for resilience to isostatic load, varies under normal production within ±5 mm. This, together with the quality of the different inspection procedures during manufacturing and the final ultrasonic inspection, means that the probability of exceeding the allowed variation of ±10 mm can be deemed as insignificant.

The above statements are based on analyses of BWR inserts because the analyses of PWR inserts are limited since the development of the manufacturing technique for PWR has been carried out on a considerably smaller scale. Consequently, the experience from the various inspections performed is limited. Therefore, the damage tolerance analysis for PWR inserts has been carried out by using material data from manufactured BWR inserts. However, since the PWR design is more robust due to the larger material thickness in the insert, it can, nevertheless, be presumed that the PWR insert will conform to the design premises.

The damage tolerance analysis of the bottom of the insert has so far only been conducted for crack-like defects and not for volumetric defects. However, the collapse loads without postulated defects for the bottom are similar to those for the cylindrical part of the insert, indicating that the acceptable volumetric defects will also be similar. This is therefore not considered as critical from the point of view of inspection.

Canister isostatic load resistance data for use in SR-Site are formally qualified in the **Data report**, based on the same sources as cited here.

Shear load in the repository

The canister shall also withstand shear loads that may occur when fractures intersecting the deposition hole experience secondary shear movement as a result of potential, large earthquakes in the vicinity of the repository. The design premises require that the copper corrosion barrier should remain intact after a 5 cm rock shear movement at 1 m/s for buffer material properties of a 2,050 kg/m^3 Ca-bentonite. This applies for all locations and angles of the shearing fracture in the deposition hole, and for temperatures down to 0°C. The insert should maintain its pressure-bearing properties to isostatic loads after such shear movements.

Regarding the fulfilment of the design premises for shear movements, the following is concluded.

- Strength calculations of the canister's resistance to shear load /Raiko et al. 2010/ verify that a canister with properties in conformity to the specification for the reference design withstands the design basis shear load.
- The material properties obtained in the test manufactured components are homogeneous and conform to the specification for the reference design, and the inspection of the material properties is performed by well-proven techniques. The results from inspections of manufactured canister components show that the specified values for fracture toughness and yield tensile strength in the manufactured series of inserts, and elongation and creep ductility in the copper shells conform to the reference design (values given in the **Canister production report**).

- According to the damage tolerance analyses the maximum acceptable depths for crack-like surface defects in the circumferential direction is 4.5 mm for semi-elliptical shape and 8.2 for semi-circular shape. The insert is less sensitive to internal defects, allowing defects larger than 10 mm. The analyses also show that the results are clearly dependent on the buffer density. A lower buffer density means that larger defects can be accepted in the insert.
- The damage tolerance analysis gives acceptable defect sizes that put rigorous requirements on manufacturing and NDT capability for the insert. Based on the results and experience so far it is expected that these additional requirements can be implemented in production and the testing methods for verification.
- The initial state for the shear load case does not take into account PWR inserts as representative materials data for strength and damage tolerance analysis are not yet available. However, the PWR design is more robust due to the higher material thickness in the cast insert.

Canister shear load resistance data for use in SR-Site are formally qualified in the **Data report**, based on the same sources as cited here.

Uneven pressure from bentonite buffer

The canister may be subjected to asymmetric loads during different phases in the repository evolution. This could temporarily occur due to uneven water saturation in the buffer. Permanent asymmetric loads may occur due to uneven density distribution of the saturated buffer due to irregularities in the geometry of the deposition holes. The resulting bending stresses in the cast iron insert have been evaluated with simplified calculations /Raiko et al. 2010/ and are lower than the yield strength.

The canister properties and design parameters that are important for loads from uneven pressure from the bentonite buffer are in principle the same as for the isostatic load case and therefore the conclusions regarding isostatic load are valid also for this case, i.e. deposited canisters that conform to the reference design will also conform to the design premise. One additional restriction, however, is based on circumferential crack-like defects in the insert. Such defects with up to 48 mm depth can be accepted. The damage tolerance analysis presented in /Raiko et al. 2010/ verifies that the canister strength is sufficient to withstand the uneven pressure from the bentonite buffer if the design parameters of importance for the isostatic load case conform to the values specified for the reference design.

In conclusion, the probability that the canister will not withstand the loads is negligible.

Corrosion load

A main function of the canister is to provide a corrosion barrier, and the key design premise is a nominal copper thickness of 5 cm, also considering the welds. The copper in the canister should be of high purity (> 99.99% Cu) and to avoid corrosion coupled to grain boundaries the oxygen content must be less than some tens of ppm.

The thicknesses of the tube, welds, lid and base according to the reference design are judged to conform, at the current level of detail of design premise, to a nominal copper thickness of 5 cm. The reference design fulfils, through its specification, the requirement on maximum oxygen content in the copper shell.

The quality of the corrosion barrier is determined by the thickness and the material composition of the copper shell. The thickness of the copper shell of a deposited canister is determined by:

- The thickness according to the reference design.
- The efficiency of the applied techniques for machining the canister's components and welds.
- The occurrences of defects induced during hot-forming and welding processes.
- The occurrence of surface damage during transportation, handling in the facilities and at deposition.

The copper thickness for the initial state is thus determined by all possible causes of reduction of the corrosion barrier. Conclusions regarding this aspect of the initial state are based on experience from test production as described in the **Canister production report** and briefly discussed in Section 5.4.2.

The design premise for the corrosion barrier is given as a nominal thickness of 5 cm. The minimum copper thickness at the initial state is given in Table 5-9. This is the manufactured and inspected thickness of the corrosion barrier after the final machining of the canister components. The initial state thickness also takes the occurrence of local internal defects in the copper shell into consideration. It is noted that the initial state deviates slightly from the design premise of a copper thickness of 5 cm. The initial state data is though a sufficient input to the further assessment of corrosion processes in SR-Site.

Defects may also occur during handling and transportation of the canister. The size of these defects has, at this stage, not been quantified.

The oxygen content in the major part of the copper shell is well below some tens of ppm, the exception being the welds produced in air where higher values have been measured in the overlapping parts. Based on this further investigations are needed to determine if the welding process will be performed in air or in argon gas.

Storing the copper canisters for extended periods of time before disposal will have a negligible effect on their service life after disposal. The total corrosion attack even after two years' storage is expected to be less that 1 μm, see the **Fuel and canister process report**, Section 3.5.4.

Copper thickness data for use in SR-Site are formally qualified in the **Data report**, based on the same sources as quoted here.

Criticality

The canister shall prevent criticality and in the analysis of the propensity for criticality of the encapsulated spent fuel assemblies for the reference canister design, see Section 5.3.4 and /SKBdoc 1193244/ for details, it is shown that the highest reactivity occurs when the assemblies are located close together towards the centre of the canister. The distance between the channel tubes (centre to centre, C-C, distance) of 210 mm for the BWR insert and 370 mm for the PWR insert is specified in the reference design. Hereby, the reference design conforms to the design premises in criticality respect. However, so far, no verification of the C-C distance between compartments by physical measurement has been done on manufactured inserts. The verification parameter related to the C-C distance has yet to be determined.

The calculations also show that of the elements occurring in nodular iron, silicon (Si) and carbon (C), are more potent neutron reflectors than iron. The content of these substances shall be kept below 6% (Si) and 4.5% (C). Analyses of the material compositions of the nodular cast iron used for the five serial-manufactured BWR inserts and three PWR inserts show the carbon content is below 4.5% (by at least a factor 1.2) and the silicon content is below 6% (at least by a factor of 2.5) in all manufactured inserts.

Material composition and structure, and the environment within the sealed canister

The **Canister production report** shows that these design premises with regard to the material composition and structure, and on the environment within the sealed canister, will be fulfilled for the inspected canisters leaving the encapsulation plant. In summary:

- The maximum copper content of < 0.05% in the insert is verified by conventional material analysis during production. This will guarantee the conformity to this design premise.
- The material composition in the copper shell regarding content of sulphur, phosphorus and hydrogen is verified by conventional material analysis during production. The destructive inspections during manufacturing and the following ultrasonic inspection verify that the average grain size in the copper shell will be below 360 μm. This will guarantee the conformity to this design premise.
- The design of the system for drying the fuel and change of atmosphere in the insert in the encapsulation plant will guarantee that the maximum water content will conform to the design premise.
- The temperature on the surface of the copper shell must not be substantially above 100°C, which has to be considered in the instructions for handling the canister in the facilities and during transportation as well as in the detailed design of the canister transport cask. As the temperature is a parameter that is relatively trivial to measure, the conformity to this design premise can be verified.

- The radiation dose rate at the canister surface has been calculated in /SKBdoc 1077122/. The highest obtained radiation dose rate was 0.18 Gy/h on limited areas of the canister. Since the decay power and the radiation are related to the radioactivity of the assemblies, it can be concluded that the radiation dose rate on the canister surface will be well below the acceptable 1.0 Gy/h as long as the fuel assemblies selected for encapsulation conform to the decay power criterion.
- Residual stresses induced in the material during manufacturing processes like casting, hot-deformation, welding or machining, are secondary stresses. They do not have any external driving force that would continue their existence after yielding or thermal stress relief treatment of the material. Measurements of residual stresses in inserts show that these have no practical influence on limit load or other higher loads that cause yielding, because the manufacturing-based residual stresses are expected to vanish when the material yields (see Section 7.3.2 in /Raiko et al. 2010/). The interest in residual stresses in copper shells is primarily related to the possibility that stress corrosion cracking might appear for unforeseen reasons or that elastic deformation might take place. For analyzing the potential risk for stress corrosion a comparison between primary (after creep) and residual stresses could be done. It can be concluded (Section 6.2.10 in /Raiko et al. 2010/ that the primary stresses are more important for analyzing stress corrosion cracking than the secondary residual stresses.

5.5 Initial state of the buffer

The **Buffer production report** presents the design premises, the reference design, verifying analyses that the reference design does fulfil the design premises, the production and control procedures selected to achieve the reference design, verifying analyses that these procedures do achieve the reference design and an account of the achieved initial state. The following sections give a summary of the contents of the **Buffer production report** and in particular of the initial state.

5.5.1 Design premises relating to long-term safety

The main function of the buffer is to restrict water flow around the canister. This is achieved by a low hydraulic conductivity, which makes diffusion the dominant transport mechanism, and a swelling pressure, which makes the buffer self sealing. The buffer should also keep the canister in position in the deposition hole, damp rock shear movements and maintain its properties for the timescale of the assessment. The buffer should, furthermore, limit microbial activity on the canister surface and filter colloidal particles. The buffer should not significantly impair the functions of the other barriers. Quantitative design premises are provided in /SKB 2009a/ and are outlined as follows.

After swelling,

- The swelling pressure should exceed 2 MPa.
- The hydraulic conductivity should not exceed 10^{-12} m/s independently of dominating cation and for chloride concentrations up to 1 M.
- The shear strength must not exceed the strength used in the verifying analysis of the canister's resistance against shear loads. These conditions apply for temperatures down to 0°C and temperatures up to 100°C.

The above requirements are fulfilled for a saturated buffer that has:

- A density in the interval 1,950–2,050 kg/m^3.
- A montmorillonite content corresponding to a dry buffer material containing 75–90 percent by weight.

The relation between montmorillonite content and the requirements on the buffer material is discussed in /Karnland 2010/. Regarding material composition, it is also required that:

- The content of organic carbon should be less than 1 wt %.
- The sulphide content should not exceed 0.5 weight percent of the total mass, corresponding to approximately 1% of pyrite.
- The total sulphur content (including the sulphide) should not exceed 1 wt-%.

The water saturation and swelling processes form part of the long-term evolution of the buffer and cannot be inspected at the initial state. Rather, based on analyses of these processes, design premises for the geometry and the density at the initial state can be stated and a reference design conforming to the design premises derived and specified.

5.5.2 Reference design and production procedures

Reference design

The reference design of the buffer is described by a set of *design parameters* for which nominal values and acceptable variations are given. The design parameters will be inspected in the production to confirm that the produced buffer at the initial state conforms to the reference design and to provide an estimate of the actual properties of the buffer at the initial state.

The reference design of the buffer consists of specifications of:

- The material composition.
- The material ready for compaction.
- The fabricated blocks and pellets.
- The installed buffer.

The reference buffer geometry is presented in Figure 5-11. The buffer consists of one solid bottom block, six ring-shaped blocks around the canister and three solid blocks on top of the canister. The buffer ends and the backfill commences at the top of the third block on top of the canister. The centre line of the buffer blocks coincides with the centre line of the deposition hole. The gap between the blocks and the rock surface of the deposition hole is filled with pellets. The thickness around the canister will, for the installed buffer, deviate from the nominal thickness, i.e. 35 cm. The installed buffer thickness will depend on the diameter of the deposition hole and its variation along the hole and on the position of the ring-shaped blocks within it. The buffer thickness will also be affected by the position of the canister within the ring shaped blocks and the diameter of the canister. The canister will be guided so that it is placed centred within the buffer ring.

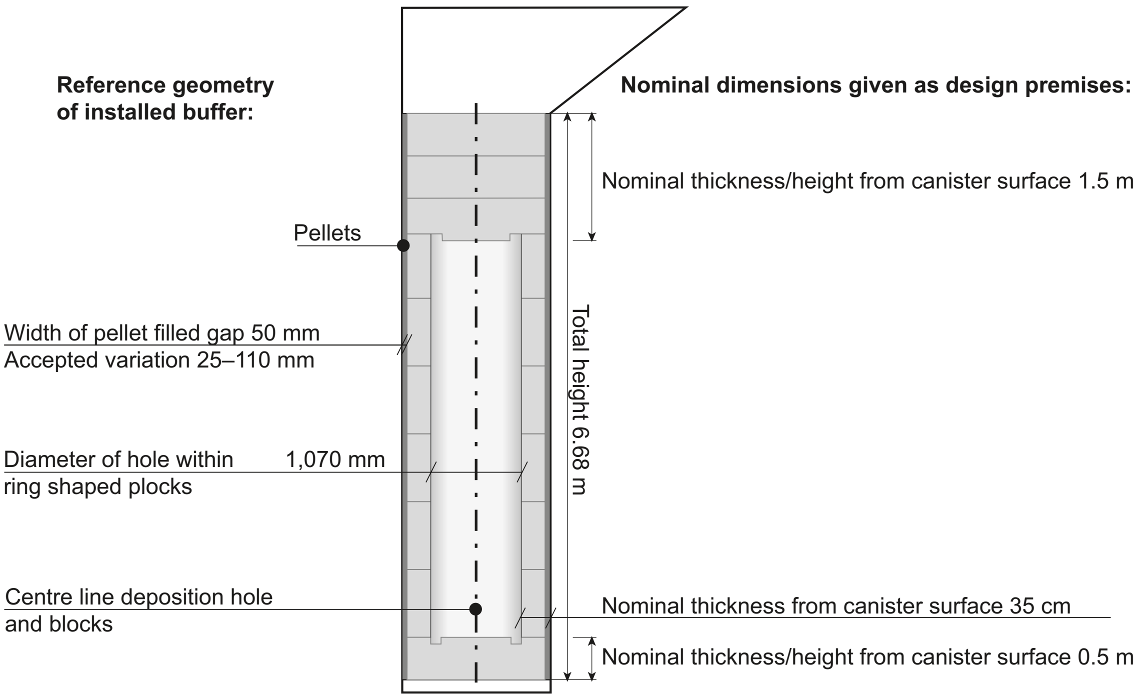


Figure 5-11. *Reference geometry of the installed buffer, see the **Buffer production report**, Figure 3-3.*

The reference buffer material is a bentonite clay which fulfils two basic mineralogical criteria: Firstly, the montmorillonite content has to be sufficiently high to uphold and maintain the minimum swelling pressure, maximum hydraulic conductivity and allowed stiffness and shear strength for the specified density interval and secondly, the content of harmful accessory minerals has to be low.

In SR-Site two example materials that conform to the specification given in Table 5-10 are assessed. The examples, MX-80 and Ibeco RWC (called Deponit CA-N in SR-Can) are both from large deposits and are mined by large bentonite suppliers. They are of different origin and do have different dominant counter-ions and should be seen as relevant illustrations of possible alternatives to be used in the repository.

The density and homogeneity of the produced blocks and pellets will depend on the granule size distribution and water content of the material to be compacted and on the compaction pressure. To achieve high reliability in the production, the granule size distribution and water content must be specified. The reference designs of the blocks and pellets are presented in Table 5-11. The densities are given as bulk densities since it is the bulk densities that are going to be inspected in the production.

The canister lids and bottoms are not flat but contain hollows and edges, see Section 5.4.2. These volumes must be filled with bentonite. This is done by specially machined solid blocks. The buffer takes up additional water after installation and will develop a swelling pressure at the end of the water saturation process. This will lead to an upward expansion of the buffer and a corresponding compaction of the overlying deposition tunnel backfill until an equilibrium is reached. The reference design, in particular the reference density, has been determined with regard to this swelling process, so that the required final density of 1,950–2,050 kg/m^3 is obtained after swelling. Such expansion calculations, verifying that the buffer reference design fulfils the design premises, are accounted for in the **Buffer production report** and are summarised briefly in Section 5.5.3. The expansion process is also part of the long-term evolution, and is further accounted for in Section 10.3.9 of this main report.

Table 5-10. Dominant cation, CEC and accessory minerals for MX-80 and Ibeco-RWC /Karnland 2010/.

Component	MX-80 R1 (wt-%)	Ibeco RWC R1 (wt-%)	Uncertainty (± wt-%)
Calcite + Siderite	0–1	6	1
Quartz	3	0–1	1
Cristobalite	0–1	0–1	1
Pyrite*	0.24	0.5	0.07
Mica	4	6	1
Gypsum	0–1	0–1	1
Feldspars	4	0–1	2
Dolomite	0	1	1
Montmorillonite	84	81	3
Na-	75%	24%	5
Ca-	16%	45%	5
Mg-	7%	29%	5
K-	2%	2%	1
Organic carbon*	0.24	0.23	0.05
CEC (meq/100g)	75	70	3

*Pyrite and organic content are based on chemical analyses and not on XRD data.

Table 5-11. Reference buffer blocks and pellets.

Design parameter	Nominal design	Accepted variation
Solid bocks		
Bulk density (kg/m^3)	2,000	±20
Water content	As in the material ready for compaction.	As in the material ready for compaction.
Dimensions (mm)	Height: 500 Outer diameter: 1,650	±1
Ring-shaped blocks		
Bulk density	2,070	±20
Water content	17 (As in the material ready for compaction.)	±1 (As in the material ready for compaction.)
Dimensions (mm)	Height: 800 Height of top block: 760 Outer diameter: 1,650 Inner diameter: 1,070	±1
Pellets		
Dimensions (mm)	16·16·8	–
Bulk density loose filling (kg/m^3)	1,035	±40
Water content	17 (As in the material ready for compaction.)	±1 (As in the material ready for compaction.)

Production handling and installation

The production line for the buffer consists of three main parts:

- Excavation and delivery.
- Manufacturing of blocks and pellets.
- Handling and installation.

Bentonite deposits exist at many places around the world and excavation and delivery can be made by alternative companies. The desired material properties will be specified at ordering. Each shipment of bentonite will be accompanied by a protocol from the supplier that describes the actual composition of the delivered material. The delivered material is inspected as a basis for the acceptance of the delivery. Inspections for the production comprise measurement of water content and granule size distribution of the delivered material and measurement of its total weight. Design parameters of importance for the functions of the buffer to be inspected comprise mineralogical composition of the buffer material. The mineralogical composition of the bulk material is analysed as random powders by a standard X-ray diffractometer (XRD). The mineral distribution is evaluated by use of XRD quantitative software based on the Rietveld refinement method, see the **Buffer production report**. Specific analyses are made for sulphur and carbon.

Before compaction, the water content of the material is adjusted to suitable values for the process. The reference method for pressing of blocks is uniaxial compaction.

After compaction, the weight and dimensions (height and (inner and) outer diameter) of each block are inspected. The dry density of the block is calculated from the recorded water content, weight and dimensions. The blocks are machined to specified dimensions in order to achieve a well-aligned stack of blocks in the deposition hole and the required installed bulk density. The reference method to manufacture pellets is to compact the conditioned material to small pellets.

A statistical evaluation of the blocks manufactured for the Prototype Repository shows that the block densities are normally distributed, see the **Buffer production report**. As an estimation of the density distribution resulting from a manufacturing process adapted to the selected material, the density of a total of 25 blocks (10 ring-shaped and 15 solid blocks) made from the same delivery of bentonite and compacted on the same occasion has been analysed. The resulting densities are presented in Table 5-12. The densities presented in Table 5-12 are, based on current experience, the variation in block densities that can be expected from a manufacturing process adapted to the delivered material.

Table 5-12. Expected bulk densities of blocks based on measurements of blocks made from the same delivery of bentonite and compacted on the same occasion. The total number of blocks used in the evaluation is 25, 10 ring-shaped blocks and 15 solid blocks.

Bulk density of ring-shaped blocks (kg/m^3)				Bulk density of solid blocks (kg/m^3)			
Mean	Std	99.9% C.I.		Mean	std	99.9% C.I.	
2,070	5.81	2,050	2,088	1,998	5.79	1,978	2,017

The installation of the buffer is based on the fact that deposition holes, according to specifications, are provided from the underground opening production line. The excavation and inspection of deposition holes is described in the **Underground openings construction report**.

Experience from the Prototype Repository shows that the diameter of the deposition hole is normally distributed, see the **Buffer production report**. Based on three measurements of the diameter on each level with an equidistance of 0.4 m in depth the average diameter was determined to be 1.760 m with a standard deviation of 2.02 mm. In the Prototype Repository the configuration of the drilling equipment was set to a larger nominal diameter than 1.750 m. When describing the statistical variability of the dimensions of a deposition hole drilled in the final repository it is therefore reasonable to assume a mean value of the diameter of 1.750 m and, based on current experience, to assume a standard deviation of 2.02 mm. At Äspö HRL the deviation between the centre points of the cross section measured every 40 cm and a line connecting the centre points of the section in the tunnel floor and bottom of the deposition hole has been measured. The deviation is in the order ±10 mm, see the **Buffer production report**, Section 6.1.4. This can be regarded as an estimation of the straightness of the holes and the corresponding variation in width of the pellet-filled gap. Spalling will result in a local increase in the deposition hole radius as illustrated in Figure 5-12. As stated in Section 5.2.2, the potential depth of spalling has been estimated as part of the repository design work, see Figure 5-5.

To protect the buffer from taking up water from the moisture in the air and from drying if the deposition hole is dry, a protection sheet is placed in the deposition hole before installation of the blocks.

The bottom block is installed and centred with respect to the average centre line of the deposition hole. The first ring shaped block is installed in the same way as the bottom block. The installed ring shaped block is used to guide the installation of the next ring shaped block so that they are positioned in the same horizontal position on top of each other with a straight and centred hole for deposition of the canister in the middle. After installation of the top ring shaped block the hole within the blocks is inspected

The canister can then be deposited as described in the **Canister production report**.

When the canister has been deposited the three top buffer blocks and the two blocks considered as a part of the backfill, see Section 5.6, are installed. From the dimensions of the blocks and their positions the buffer thickness and block filled part of the deposition hole volume can be calculated. The position of the blocks is together with their weights and the deposition hole dimensions used as input to calculate the bulk density of the installed buffer and its variation in the deposition hole. Before the installation of the pellets the protection sheet and the sensors placed within the sheet are removed.

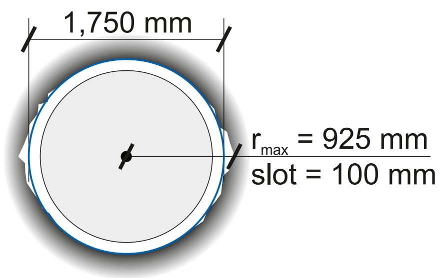


Figure 5-12. Increase in deposition hole radius due to spalling and rock fall out in a deposition hole with nominal diameter.

The pellets are filled into the gap by placing a conical hood on top of the last installed bentonite block and pouring the pellets into the deposition hole. As soon as the gap has been filled with pellets the backfilling of the deposition tunnel continues so that the backfill can prevent swelling and expansion of the buffer caused by a fast water uptake of the pellet filling. Installing the blocks according to Table 5-12 in a deposition hole with an average diameter of 1.750 and standard deviation 2.02 mm results in the calculated installed density presented in Table 5-13. In the calculations the variations in water content and bulk density of the pellet filling are neglected. The bulk density of the pellets is set to 1,035 kg/m³ and the water content in blocks and pellets is set to 17%.

If the deposition hole is not straight the centre point of each individual cross section will not coincide with the average vertical centre line of the hole, see Figure 5-13. As a result buffer blocks will not be placed centred in each individual cross section and the width of the pellet-filled gap will vary over the cross section.

Table 5-13. Results of calculations of installed buffer density for different sections in a deposition hole.

Section around the canister				Section above and below the canister			
Dry density, ρ_d (kg/m³)		Resulting saturated density, ρ_m (kg/m³)		Dry density, ρ_d (kg/m³)		Resulting saturated density, ρ_m (kg/m³)	
Mean	Std	99.9% C.I.		mean	Std	99.9% C.I.	
1,577	4.72	2,000	2,020	1,616	4.73	2,025	2,045

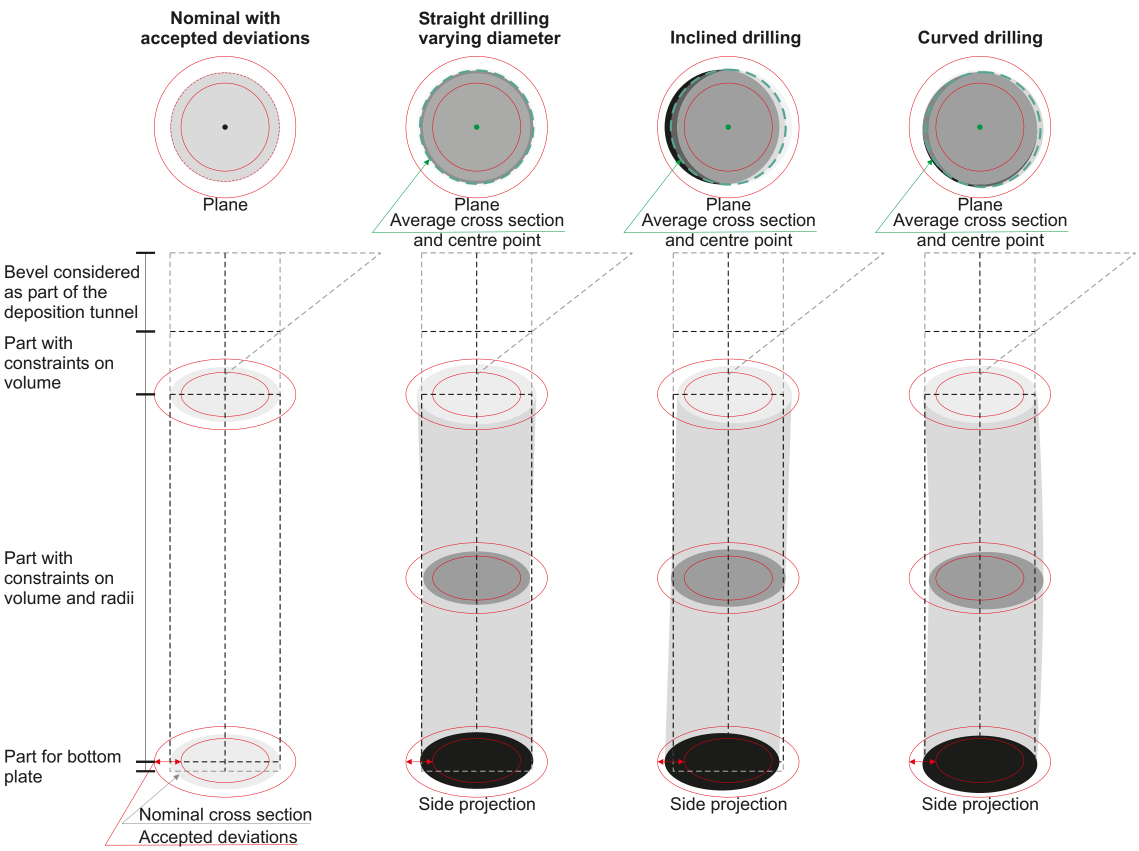


Figure 5-13. Some possible causes of deviations in the geometry in a deposition hole without rock fall out. All kinds of deviations can occur in the same deposition hole. The deviations are exaggerated in scale. (The bevel is discussed in the backfilling section).

The techniques for loading, transportation and storage of the buffer blocks and pellets are well known from similar industrial applications. SKB has such experience from the Äspö HRL. Blocks have been stored placed on pallets with an air tight hood for several months without showing any changes in weight or water content. The function of the protection sheet placed in the deposition hole before installation of the buffer has been tested at Äspö HRL. The technique to fill the outer gap with pellets has been used in several large scale tests at Äspö HRL with good results.

5.5.3 Initial state

The initial state of the buffer is the state when the auxiliary equipment used during installation is removed and all buffer components are installed in the deposition hole. Inflow of groundwater to the deposition hole and its impact on the buffer is not accounted for in the initial state.

The properties of the buffer to be designed to conform to the design premises for long-term safety are:

- Material composition.
- Installed density.
- Installed geometry.

These properties are to some extent interdependent. The design density, for example, is based on a given material composition. In the **Buffer, backfill and closure process report**, the buffer is characterised by a number of variables. Most of the initial state values for these variables are determined by the design properties. This is illustrated in Table 5-14.

Material composition

The reference design is based on a bentonite with a given montmorillonite content and a maximum inventory of certain impurities (see Section 5.5.1).

Regarding design premises relating to material composition, it is concluded that the required montmorillonite content, and the specified limitations on organic carbon, sulphide and sulphur are fulfilled at the initial state. This conclusion is based on the presented analyses of the compositions of the two example materials and of the handling and inspection procedures outlined in the description of the production of the buffer.

Table 5-14. Relation between the designed buffer properties and the variables used in the safety assessment. References to where, or how, initial state values of the variables not related designed buffer properties can be found or derived.

Variable	Buffer property	Initial state values
Water content	Material composition	
Gas content		
Bentonite composition		
Montmorillonite composition		
Porewater composition		
Hydrovariables (pressure and flows)	Material composition	
Stress state	Installed density	
Pore geometry	Installed density	
Buffer geometry	Installed dimensions and geometrical configuration	
Radiation intensity	–	Calculated Radiation intensity on canister surface **Spent fuel report**, Section 6.6.
Temperature	–	Calculated
Structural and stray materials	–	Bottom plate material and dimensions according to the reference design **Underground openings construction report**, Figure 5-3.

To evaluate the long term performance of the buffer a more detailed characterization of the material is needed. Important parameters in the description of the materials are:

- Chemical composition.
- Mineralogical composition.
- Grain density.
- Specific surface area.
- Grain size distribution.
- Water content.

In addition to this the clay fraction of the material is characterized by:

- Structural formula including:
  - Layer charge.
  - Charge distribution.
- Cation exchange capacity.
- Original exchangeable cations.
- Charge distribution.

These parameters are further described in the **Buffer process report** and are defined as the Bentonite composition and the Montmorillonite composition variables.

In natural bentonite, the charge compensating cations are rarely of one element alone, but a mixture of both mono- and divalent ions. The swelling properties are to a large extent dependant on the magnitude and the position of the layer charge, but also on the type of charge compensating cation. The dominating cation is therefore often used to describe the type of bentonite, e.g. sodium bentonite, although the content of other ions may be quite large. High-quality commercial bentonites normally contain over 80% of montmorillonite, which is expected to give various bentonite products similar sealing properties. However, the other minerals in bentonite may vary substantially within, and especially between, different quarries. Typical accessory minerals are other clays, feldspars, quartz, cristobalite, gypsum, calcite and pyrite. There are quantitative limits for sulphide, total sulphur and organic carbon, see 5.5.1. No quantitative criteria for other accessory minerals in the bentonite are defined at this stage. The mineralogical composition of the materials used for SR-Site is presented in Table 5-10.

For the purpose of the calculation of radiation shielding as well as the check on the maximum contents of sulphur and organic carbon it is important to determine the total chemical composition of the bentonite. The mean chemical composition of the bulk materials from Table 5-10 expressed as oxides are for MX-80: 57% SiO_2, 18.5% Al_2O_3, 3.6% Fe_2O_3, 2.3% MgO, 1.3% CaO, 2.0% Na_2O, 0.5% K_2O, 0.2% TiO_2, 0.3% total carbon, 0.3% total sulphur, and 13.7% loss on ignition and for Ibeco-RWC it is 48% SiO_2, 16% Al_2O_3, 4.6% Fe_2O_3, 2.9% MgO, 5.4% CaO, 0.7% Na_2O, 0.8% K_2O, 0.7% TiO_2, 1.0% total carbon, 0.7% total sulphur, and 20% loss on ignition.

The initial conditions of all other parameters for bentonite and montmorillonite composition, together with their relevance for the long term performance, are discussed in /Karnland 2010/.

Water content, gas content and porewater composition

There are no specific design premises with regard to water content, gas content and porewater composition, but these properties need to be known for the subsequent analysis. In the reference design, the initial water content in blocks and pellets is selected to be 17%. Water content is determined by a standard geotechnical method. The original water content in the buffer material is adjusted to facilitate the manufacturing process. All porosity not filled with water contains air. The initial porewater composition may be calculated but not directly measured.

Installed density

The design density range of 1,950–2,050 kg/m^3 is set for a fully saturated and homogenized buffer in the deposition hole. The initial state on the other hand represents the installed buffer blocks and pellets with a water content given by the manufacturing process. The parameters in the initial state should produce a buffer that lies within the density range for every cross-section in the deposition hole, neglecting the effects of incomplete homogenization. Analyses, presented in the **Buffer production report**, are made to show how to fulfil the design premise on buffer density.

The installed buffer density will depend on the density and dimensions of the installed blocks and pellets, i.e. the installed buffer mass, and the volume of the deposition hole and canister. The impact of the variations of volume of the canister and dimensions of the blocks on the installed buffer density can be neglected. The important parameters are the density of the blocks and pellets and the volume of the deposition hole. Over a random cross section of the deposition hole the installed density will vary due to:

- Variations of the diameter of the deposition hole.
- The placement of the buffer blocks with respect to the centre line of the deposition hole.
- The occurrence of spalling.

The impact on the variation in deposition hole diameter on the installed and corresponding saturated density can be determined assuming that the canister and ring shaped blocks are placed centred in each cross section. This will be the case if the centre point of each cross section is coincident with a vertical line, i.e. the drilling is straight.

Based on the distribution in block densities presented in Table 5-12, the variation in saturated density can be calculated for allowed increases in width of the pellet filled gap, or deposition hole radius, from the nominal. Results from such calculations are presented in Table 5-15. For a straight deposition hole with nominal diameter, the increase in width of the pellet filled gap corresponds to the maximum allowed depth of spalling. The number of deposition holes where this depth of spalling may occur as assessed when developing the current repository design /SKB 2009b/ (see also Figure 5-5) is also accounted for in Table 5-15. If the diameter deviates from the nominal the allowed depth of spalling is altered accordingly. If the deposition hole is not straight the allowed spalling decreases accordingly.

It is thus concluded that the methods for producing the buffer will yield densities at initial state that i) fulfil the specification of the reference design and ii) lead to densities after saturation that conform to the design premise on saturated density. Also, based on the results presented in Table 5-13, the densities after saturation without spalling are, with 99.9% confidence, expected to lie in the interval 2,000–2,020 kg/m^3 for sections around the canister and in the interval 2,025–2,045 kg/m^3 for sections above and below the canister. The former interval, which is narrower than that of the design premise, is of particular importance for the assessment of shear loads on the canister where a lower density yields lower shear loads on the canister. It is thus concluded that an upper limit on buffer density of 2,020 kg/m^3 for sections around the canister can be used in assessments of shear loads on the canister.

Installed geometry

The impact of the variation in canister placement and canister diameter on the buffer thickness can be neglected. The actual deposition hole diameter will deviate from the nominal. As described in the production section, analysis of measurements of diameters from the Prototype Repository shows that the standard deviation is 2.025 mm. The 99.9% confidence interval of the deposition hole diameter is 1. 743 < ø < 1.757 m. Assuming that the canister is placed in the centre of the deposition hole and there is no spalling results in a 99.9% confidence interval for the buffer thickness of 34.7 < buffer thickness < 35.3 cm.

Table 5-15. Results of calculations of the saturated buffer density at the canister sections for allowed increases in width of the pellet filled gap from the nominal.

Allowed increase in width of pellet filled gap (m)	Number of deposition holes out of 6,000 with corresponding depth of spalling	99.9% confidence interval for the buffer density at saturation (kg/m^3)	(kg/m^3)
0.050	150	1,933	1,948
0.040	200	1,946	1,961
0.030	400	1,959	1,974
0.020	600	1,972	1,988
0.010	950	1,986	2,003

Conformity of the reference design to design premises regarding swelling pressure and hydraulic conductivity

The reference design is based on a given material composition and an installed density of that material. Above, it has been demonstrated that the design premise on material composition is fulfilled at the initial state and that the initially installed density yields a density after saturation and swelling that conforms to the design premises. In the **Buffer production report** it is demonstrated that a buffer with this latter density meets the design premises regarding swelling pressure and hydraulic conductivity. For example, Figure 5-14 shows the swelling pressure for the MX-80 and Ibeco RWC (Deponit CA-N) materials exposed to NaCl and $CaCl_2$ solutions, respectively. The swelling pressure for the reference density will be 7–8 MPa for both materials. With account taken for the allowed variations in density, the swelling pressure may vary between 4.5 and 13 MPa, i.e. the design premise requiring a minimum swelling pressure of 2 MPa is fulfilled. See the **Buffer production report** and /Karnland et al. 2006/ for further details.

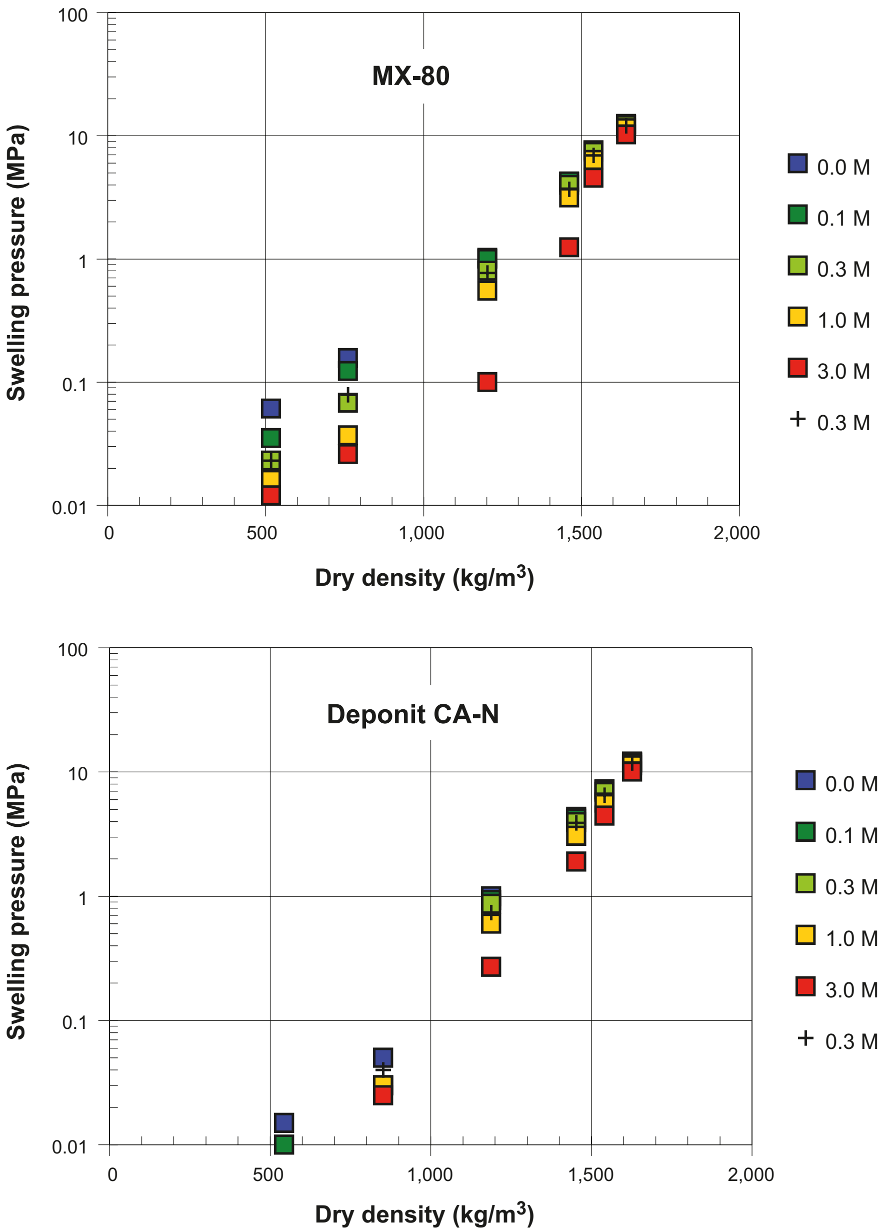


Figure 5-14. *Swelling pressures of MX-80 exposed to NaCl solutions (upper) and Ibeco RWC (Deponit CA-N) exposed to $CaCl_2$ solutions (lower). Concentration expressed as M.*

5.6 Initial state of the deposition tunnel backfill

The **Backfill production report** presents the design premises, the reference design, verifying analyses that the reference design does fulfil the design premises, the production and control procedures selected to achieve the reference design, verifying analyses that these procedures do achieve the reference design and an account of the achieved initial state. The following sections give a summary of the contents of the **Backfill production report** and in particular of the Initial state.

5.6.1 Design premises relating to long-term safety

The deposition tunnel backfill is the material installed in deposition tunnels to fill them. The purpose and function of the backfill in deposition tunnels is to keep the buffer in place and to restrict groundwater flow through the deposition tunnels.

In SR-Site, all the tunnels at the repository level as well the ramp and shaft up to the level where top seal starts are assumed to be filled with "tunnel backfill".

Design premises on the backfill related to long term safety as stated in /SKB 2009a/, are:

- Limit advective transport in deposition tunnels, which is achieved if the hydraulic conductivity $< 10^{-10}$ m/s and the swelling pressure > 0.1 MPa.
- Restrict upwards buffer swelling/expansion, which is achieved if packing and density of the backfill, both in the initial dry state and after complete water saturation is sufficient to ensure a compressibility that results in a minimum buffer saturated density around the canister according to the conditions set out (i.e. 1,950 kg/m^3) with sufficient margin to loss of backfill[11] and uncertainties.
- The backfill material does not contain substances that may cause harmful buffer degradation or canister corrosion.
- The density and material composition shall be such that the barrier functions of the backfill can be maintained over a long time.

Furthermore, the **Backfill production report** also states the general requirement that the backfill should not significantly impair the barrier functions of the barriers. Except for the design premises related to the backfill's long-term functions there are currently no design premises set for the backfill by the other barriers.

5.6.2 Reference design and production procedures

Reference design

The reference design of the installed backfill is presented in Figures 5-15 and 5-16.

The reference backfill material is a bentonite clay with the montmorillonite content of 50–60% (the accepted variation is 45–90%), i.e. less strict than what is required for the buffer. In SR-Site the bentonite called Milos Backfill (*Milos BF 04*) in /Olsson and Karnland 2009/ is used as an example of such a material. The composition of this material is provided in Table 5-16.

The reference designs of blocks, pellets and bottom bed material are given in Table 5-17. The design parameters specify the properties that the different components shall have when they are installed. The different backfill components have different densities. With respect to this, in order to facilitate the determination of the installed density in the deposition tunnel, the densities are expressed as dry density and the water content is specified.

The installed dry density will depend on the volume of the deposition tunnel and the mass of backfill material installed in the tunnel. The installed density is calculated per blasting round. It is determined from the acceptable blasting round volume, the dry densities of blocks and pellets and the portions of the volume filled with blocks, pellets and bottom bed. The larger the portion of the tunnel that is filled, and the less the void volume, the larger is the installed density.

The calculated installed density for the reference design of backfill components according to Table 5-17 and the installed backfill according to Table 5-18, are set out in Table 5-19.

[11] E.g. by erosive processes after plugging the deposition tunnel.

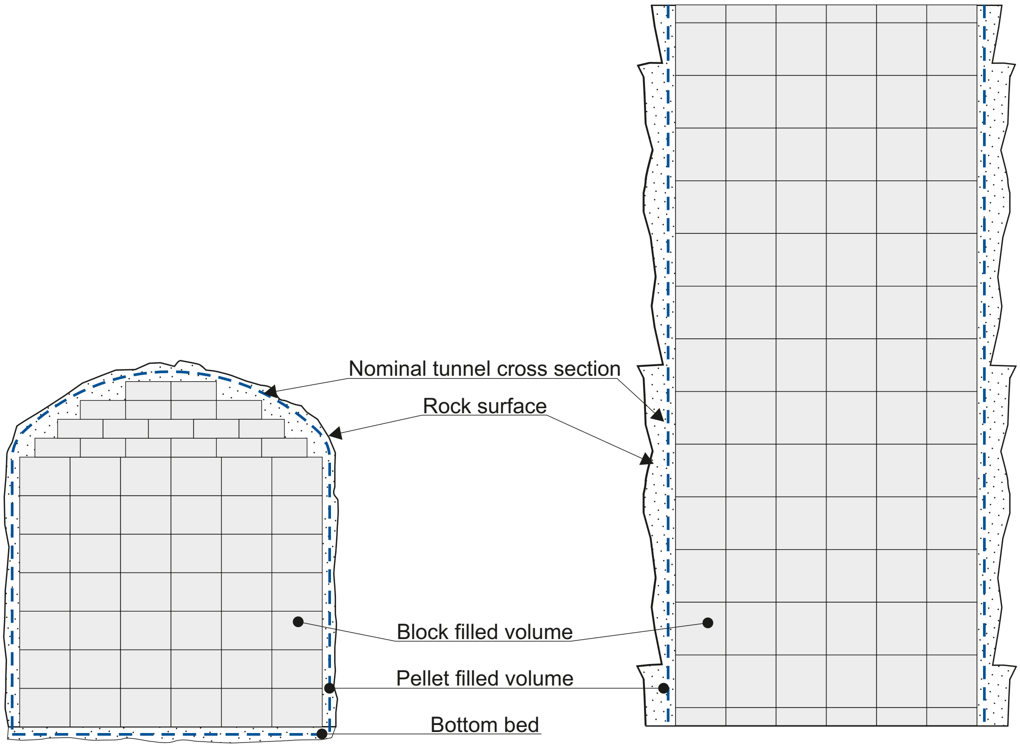


__Figure 5-15.__ Reference geometry of the installed backfill in a schematic tunnel showing vertical (left) and horizontal cross sections (revised from the __Backfill production report__).

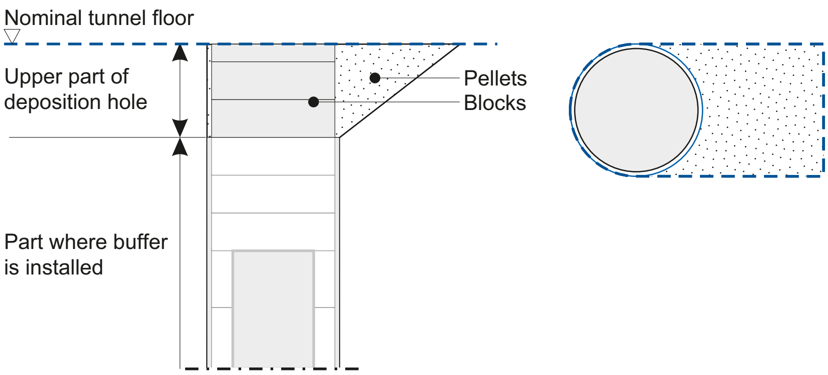


__Figure 5-16.__ Reference geometry of the backfill in the upper part of the deposition hole.

Table 5-16. Dominant cations, CEC and accessory minerals for backfill material.

Parameter	Nominal content Milos Backfill /Olsson and Karnland 2009/
Montmorillonite	58
Cations	Ca (40%), Mg (48%), Na (9%), K (3%)
CEC (meq/100 g)	73
Sulphide-S content	~0.03%
Total sulphur content (including the sulphide)	0.06%
Organic carbon	0.38%
Calcite	~8 wt-%
Mica/illite incl. Illitic layers in I/S (Illite/Smectite)	6–8 wt-%
Gypsum	~0.5 wt-%
Albite	n.d.
Ca-plagioclase	~1.5 wt-%
K-feldspar	5–6 wt-%
Dolomite	~16 wt-%

Table 5-17. Reference block, pellet and bottom bed material ready for installation (based on Milos backfill).

Design parameter	Nominal design	Accepted variation
Blocks		
Dry density (kg/m^3)	1,700	±50
Water content (%)	17	±2
	(As in the material ready for compaction)	(As in the material ready for compaction)
Dimensions (mm^3)	700·667·510	±2·2·2
	700·600·250	±2·2·2
Blocks in deposition hole bevel[1] (on top of canister)		
Dry density (kg/m^3)	1,710	±17
Water content (%)	17	±1
Dimensions (mm)	Height: 500	±1
	Diameter 1,650	
Pellets and bottom bed pellets[2]		
Dry density separate pellets (kg/m^3)	1,700	±50
Dimensions (mm^3)	~16·16·8	–
Dry density of loose filling (kg/m^3)	1,000	±100
Water content	17	±2
	(As in the material ready for compaction)	(As in the material ready for compaction)

1) In the reference design buffer blocks are used and the design parameters are the ones specified for solid blocks in the **Buffer production report**, Table 3-4.

2) In the reference design the same kind of pellets are used for the bottom bed and the gap between the blocks and tunnel walls. This may be changed.

Table 5-18. Reference design of installed backfill (based on Milos backfill).

Design parameter	Nominal design	Accepted variation
Blocks		
Block part of blast round volume[1]	According to Figure 5-15	$V_{blocks} \geq 60\%$
Free space between blocks and tunnel walls	–	Free space ≥ 10 cm
Pellet filling in gap between blocks and tunnel walls		
Pellet part of blast round volume	The volume between the installed blocks and deposition tunnel walls	–
Bottom bed		
Thickness	10 cm from nominal tunnel floor	–
Inclination perpendicular to the tunnel axis	–	< 3 mm/tunnel width
Inclination along the tunnel axis	Inclination of nominal tunnel floor	–
Dry density compacted bed	> 1,200 kg/m^3	–

1) Including blocks in the upper part of the deposition hole and excluding slots between blocks.

Table 5-19. Calculated installed dry density for the reference design of backfill components and installed backfill. The tunnel volume is set to the largest acceptable.

Parameter	Tunnel cross section	Upper part of deposition hole
Dry density of blocks	1,700 kg/m^3	1,710 kg/m^3
Dry density of pellets[1]	1,000 kg/m^3	1,200 kg/m^3
Volume fraction of slots between blocks	2%	0
Volume of pellets filling and bottom bed	Total volume minus block and void part of the volume Nominal: 25–16.8*(1+0.02) m^3/m Accepted: 25–0.60*25(1+0.02) m^3/m	2.56 m^3
Total volume	Maximum allowed tunnel blasting volume 25 m^3/m	5.17 m^3
Calculated installed dry density (nominal block part of cross section and largest acceptable tunnel volume)	1,458 kg/m^3	1,457 kg/m^3
Calculated installed dry density (acceptable block part of cross section and largest acceptable tunnel volume)	1,408 kg/m^3	–

1) For the tunnel cross section the same density is used for the pellet filling between blocks and tunnel walls and bottom bed.

Production

The production line for the backfill comprises the following three main parts:

- Excavation and delivery.
- Manufacturing of blocks, pellets and bottom bed material.
- Handling and installation.

Details about the production can be found in the **Backfill production report**.

Medium and large scale tests have been performed to test available techniques for handling of the bentonite pellets/granules and the compaction technique to get a stable bed, and also to test the performance of an installed bed i.e. when the bed was loaded with backfill blocks (settlement etc.) and there was a water flow from the rock /Wimelius and Pusch 2008/. The tests were performed with two bentonite materials, Minelco granules and Cebogel pellets. The large scale tests were performed in an artificial tunnel using concrete blocks instead of bentonite blocks since the main focus was on the installation and behaviour of the bottom bed.

5.6.3 Initial state

The initial state of the backfill is the state when the entire deposition tunnel is backfilled. Inflow of groundwater to the deposition tunnel and its impact on the backfill is not accounted for in the initial state. At this stage of development, the presented initial state of the backfill is the outcome of the design parameters that can be expected based on the experience and results from the test production.

The properties of the backfill to be designed to conform to the design premises for long-term safety are:

- Material composition.
- Installed density.
- Installed geometry.

In the **Buffer, backfill and closure process report**, the backfill is characterised by a number of variables. Most of the initial state values for these variables are determined by the design properties. The relation between the variables and the design parameters are basically the same as for the buffer (see Table 5-14).

Material composition

The material composition of the backfill material in SR-Site is given in Table 5-16.

Installed density

The density of the installed backfill blocks and pellets can be found in Table 5-20. Based on the initial state values of the design parameters of the backfill and the deposition tunnel volumes, the installed dry density, mass and porosity to be used in SR-Site have been calculated, the results are presented in Table 5-21. The reference design values are given as comparison.

Installed geometry

The geometry of the backfill is dependent on the excavated tunnel volumes. The dimensions can be found in the **Underground openings construction report**.

Conformity to design premises

Hydraulic conductivity and swelling pressure: According to the design premises the hydraulic conductivity of the backfill should be less than 10^{-10} m/s and the swelling pressure larger than 0.1 MPa. The material specific relationships between salinity and hydraulic conductivity and swelling pressure have been determined for water salt contents of 1% and 3.5% (corresponding to ocean water). The results are shown in Figures 5-18 and 5-19. The average installed dry density will for the reference design exceed 1,450 kg/m^3, which gives a considerable margin to the dry densities required according to the results in Figures 5-18 and 5-19. For higher densities the hydraulic conductivity is decreased and the swelling pressure increased. Lower salinities than the 3.5% used in the figures can be expected during long periods, see further Sections 10.3.7 and 10.4.7. For lower salinities the hydraulic conductivity will decrease and the swelling pressure increase. Consequently, the hydraulic conductivity and swelling pressure of the reference backfill material is, at the installed density, expected to conform to the design premises, and there will be a margin for density losses.

Resistance to buffer swelling: According to the design premises the backfill shall restrict upwards buffer swelling/expansion. The density and loss of buffer density by upwards swelling/expansion has been evaluated by calculations based mainly on the swelling properties of the buffer and the swelling pressure and the compressibility of the backfill. Also the influence of dry conditions in the tunnel has been investigated. An account of these calculations is given in Section 10.3.8.

Harmful effects on buffer and canister due to backfill composition: According to the design premises the backfill material must not contain substances that may cause harmful buffer degradation or canister corrosion. Currently, neither substances nor limits are given as design premises from the assessment of the long-term safety. Based on the specifications of the reference material, the impact of the backfill material composition on buffer degradation and canister corrosion is assessed in SR-Site.

Harmful effects on buffer and canister due to loss of backfill material: During operation some backfill material in already completed and plugged deposition tunnels may be lost by piping and erosion. Material may also be lost in the future during the assessment period both during and after saturation of the backfill. Neither of these material losses has yet been fully quantified. In Figure 5-20 the margins in the ratio between the installed dry density and the density needed to fulfil the design premise to restrict upwards buffer swelling/expansion (dimensioning density) for alternative degrees of block filling of the tunnel volume are shown for two materials according to the reference design, the reference material in SR-Site: Ibeco RWC-BF (Milos backfill) and another material that has been studied: Asha. As seen the density margin is 15–25% for Milos backfill and 20–33% for Asha.

Table 5-20. The backfill design parameters at the initial state.

Design parameter	Reference design	Initial state
Montmorillonite content	45–90 wt-%	45–65 wt-% (58%)
Dry density of blocks (kg/m^3)		
– tunnel section	1,700±50 kg/m^3	1,700±50 kg/m^3
– upper part of deposition hole	1,710±17 kg/m^3	1,710±17 kg/m^3
Dry density of pellet filling (kg/m^3)	1,000±100 kg/m^3	1,000±100 kg/m^3
Dry density of compacted bottom bed (kg/m^3)	> 1,200 kg/m^3	> 1,200 kg/m^3
Block part of tunnel volume	Nominal: Figure 5-15	Average: 74.1%
	Accepted: ≥60%	Min 67.3% and max 78.7%

Table 5-21. Dry density, installed mass, volume of air at the initial state of the backfill.

Parameter	Initial state	
	Average value	Range
Dry density (kg/m^3)	1,504	1,458–1,535
Mass per m tunnel (tonnes)	34.14	32.85–36.44
Porosity	0.46	0.44–0.48
Mass of water per m tunnel (tonnes)	5.80	5.56–6.20
Volume of air per m tunnel (m^3)	4.62	4.00–5.70

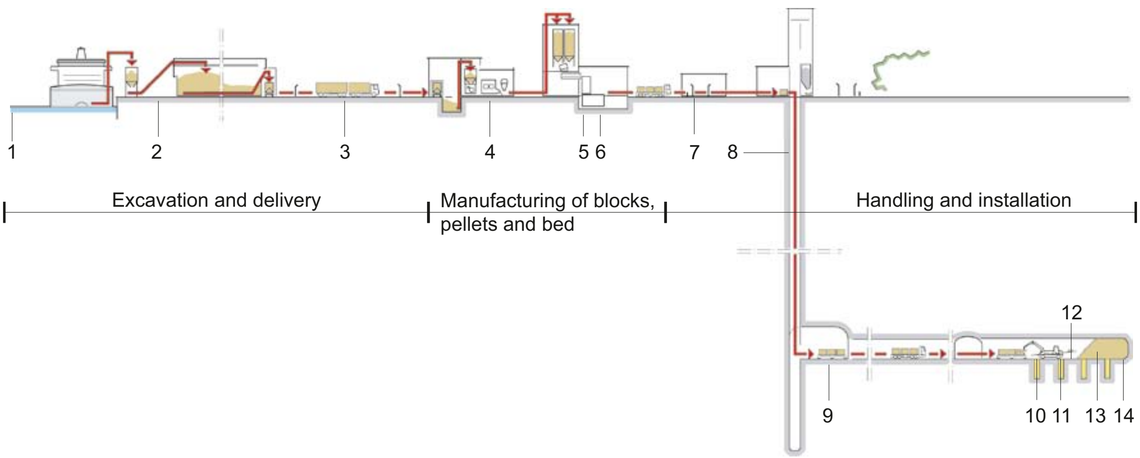


Figure 5-17. *Illustration of the backfill production line from the delivery of the material to the installation in the deposition tunnel. 1. Excavation and delivery for shipment, 2. Material delivery and intermediate storage in harbour, 3. Transport to and storage at the production plant, 4. Conditioning of the backfill material, 5. Pressing of blocks, 6. Pressing of pellets, 7. Intermediate storage at ground level, 8. Transport to and storage at repository level, 9. Preparation of deposition tunnel, 10. Installation of backfill in rejected deposition holes, 11. Installation of backfill in the upper part of the deposition hole, 12. Installation of bottom bed, 13. Installation of blocks and 14. Installation of pellets.*

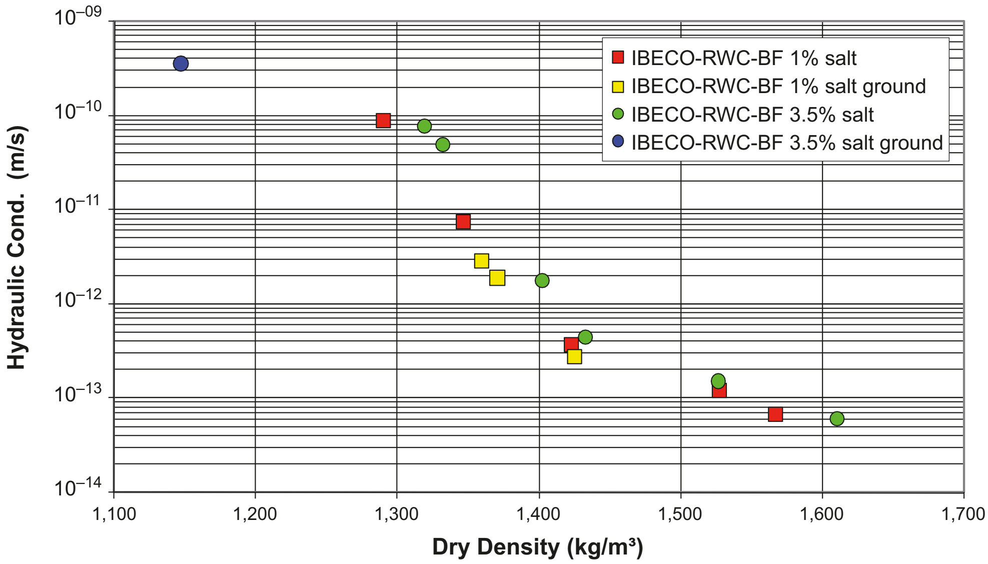


Figure 5-18. *The hydraulic conductivity of Ibeco RWC-BF as function of the dry density; "ground" refers to a material that was re-milled before compaction /Johannesson et al. 2010/.*

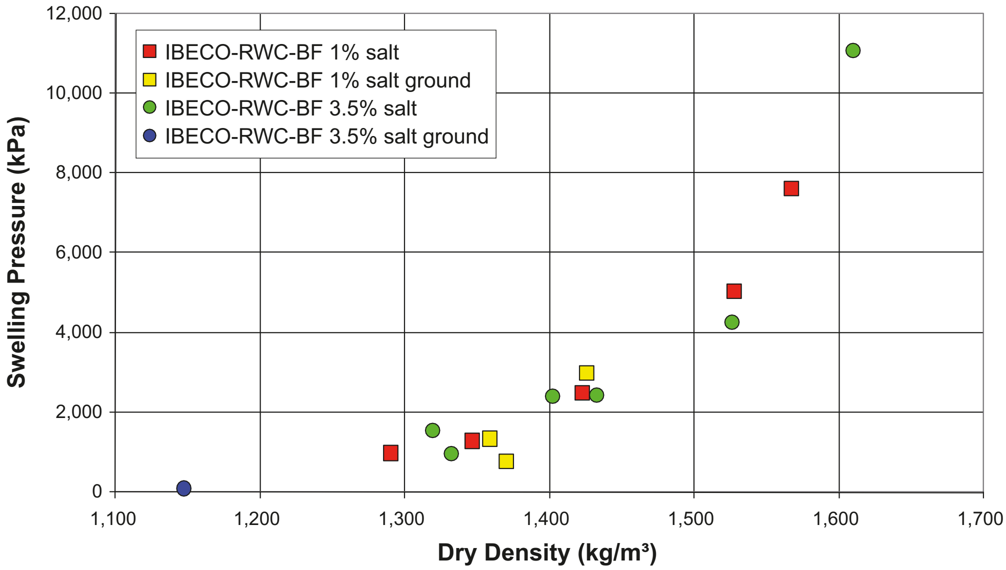


Figure 5-19. *The swelling pressure of Ibeco RWC-BF as function of the dry density; "ground" refers to a material that was re-milled before compaction /Johannesson et al. 2010/.*

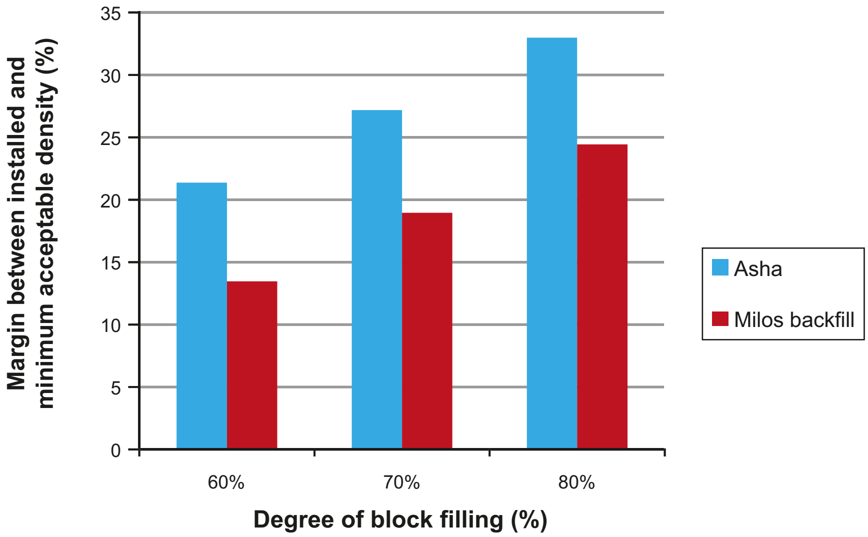


Figure 5-20. *Margin in ratio between the installed and minimum acceptable density for Milos backfill and Asha and different fractional block filling of the tunnel volume. Calculated from data in the* ***Backfill production report****, Section 4.4. The margin can be seen as the amount of backfill that could be lost without jeopardising the design premise that the backfill should ensure that the buffer density stays above its designed minimum value.*

5.7 Initial state of repository sealing and other engineered parts of the repository

This section describes the initial state of the additional engineered components in the repository. For the purpose of SR-Site, in the **Buffer, backfill and closure process report**, these are defined as:

1. Deposition tunnel plugs: presented in the **Backfill production report**.
2. Central area: presented in the **Closure production report**.
3. Top seal: presented in the **Closure production report**.
4. Bottom plate in deposition holes: presented in the **Underground openings production report**.
5. Borehole seals: presented in the **Closure production report**.

The **Closure production report** presents the reference design, production and initial states of the above components and also of:

6. Closure of main tunnels and transport tunnels.
7. Closure of ramp and shafts below the top sealing.
8. Plugs (other than deposition tunnel plugs).

However, in SR-Site the closure of all tunnels at repository level as well as the ramp and shaft below the top sealing are treated as deposition tunnel backfill, in accordance with the current reference design. Concrete used for all plugs have the same composition as the concrete in plugs in deposition tunnels.

The closure in different underground openings have different purposes e.g. to restrict groundwater flow through the underground opening, to provide mechanical restraint and to obstruct unintentional intrusion into the repository. The bottom plate in deposition holes shall provide a sufficiently flat surface for the installation of the buffer and deposition of the canister. An outline of the different kinds of closure and plugs, as provided in the **Closure production report**, is shown in Figure 5-21.

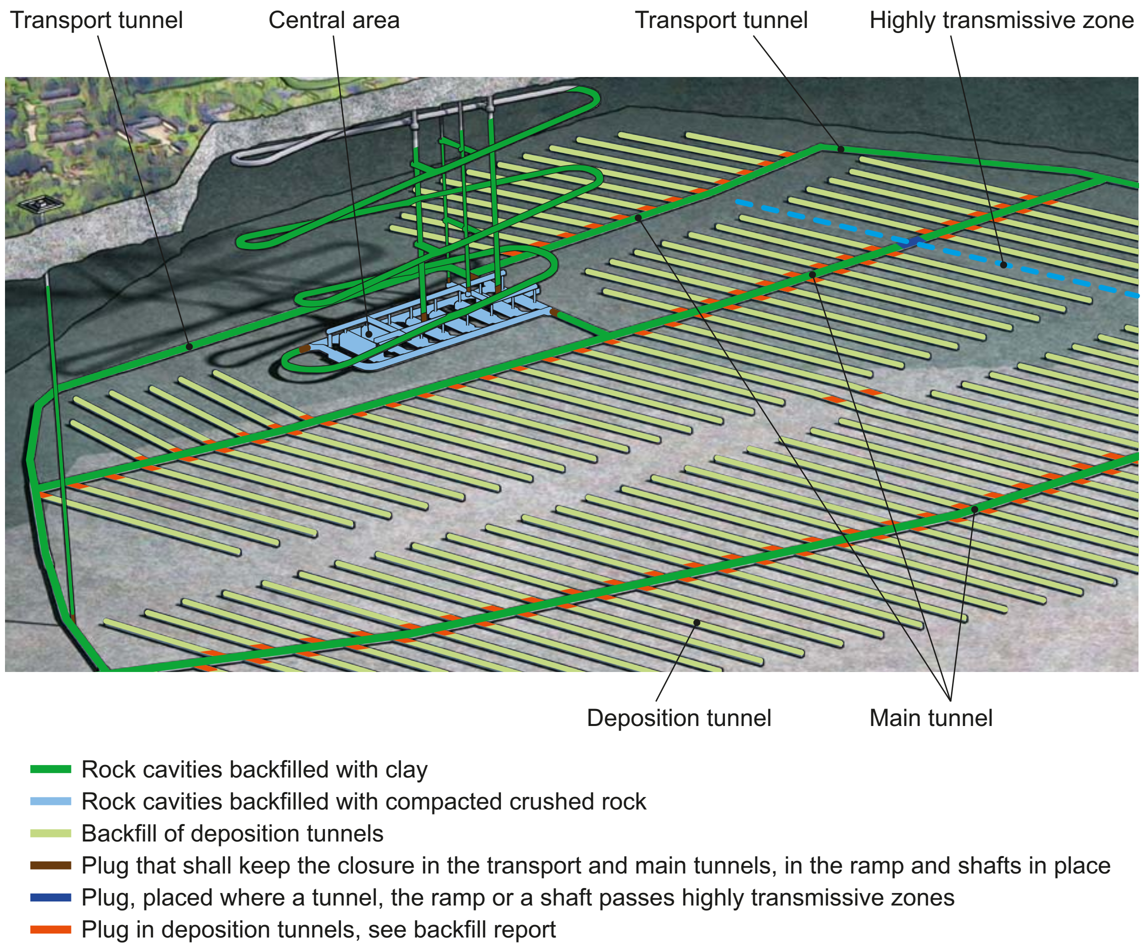


Figure 5-21. *Outline of the reference designs of closure and plugs in the different categories of underground openings (Figure 3-1 in the **Closure production report**).*

5.7.1 Design premises relating to long-term safety

Design premises have been formulated according to the following /SKB 2009a/.

- Below the location of the top seal, the integrated effective connected hydraulic conductivity of the backfill in tunnels, ramp and shafts and the EDZ surrounding them must be less than 10^{-8} m/s. This value need not be upheld in sections where e.g. the tunnel or ramp passes highly transmissive zones[12]. There is no restriction on the hydraulic conductivity in the central area.
- There is no restriction on the hydraulic conductivity in the top seal.
- The depth of the top seal can be adapted to the expected depth of permafrost during the assessment period, but must not be deeper than 100 m above repository depth.
- Only low[13] pH (< 11) materials are allowed below the level of the top seal.
- Other foreign materials must be limited – but the amounts considered in SR-Can are of no consequence.
- Boreholes must be sealed such that they do not unduly impair containment or retention properties of the repository. This is preliminary achieved if the hydraulic conductivity of the borehole seal < 10^{-8} m/s, which is ensured if the swelling pressure of the seal is > 0.1 MPa. This value need not be upheld in sections where e.g. the hole passes highly transmissive zones.

[12] As further explained in /SKB 2009a/ a short section of higher hydraulic conductivity than the ordinary closure material will have no impact on flow to and from the repository.

[13] While pH 11 is not "low" in the general sense it is low with respect to cement based materials.

5.7.2 Reference design

Deposition tunnel plug

The main function of the deposition tunnel plug is to close the deposition tunnels, keep the backfill in them in place and prevent water flow past the plug until the main tunnel has been filled and water saturated. These plugs are especially designed with respect to the properties and function of the buffer and backfill. The detailed design of the plug is under development but the water tight seal of highly compacted bentonite will be installed into a slot deepened from the excavated tunnel contour with a non-damaging technique so deep that all possible flow paths caused by excavation disturbance are cut off.

The plug in deposition tunnels consists of several parts that in different ways will contribute to maintaining its functions during the curing phase, the sealing phase and the post-closure phase of its lifetime, see the **Backfill production report**. The parts of the reference plug are illustrated in Figure 5-22.

Other plugs

Other plugs in the repository can have different purposes and are not contributory to safety. Only deposition tunnel plugs are treated specifically in SR-Site.

Central area

The only function of the closure of the cavities in the central area, see Figure 5-3, is to occupy the space with no other design premise than to prevent substantial convergence and subsidence of the surrounding rock. With respect to this the reference design is to use crushed blasted rock that will be placed in horizontal layers and then compacted.

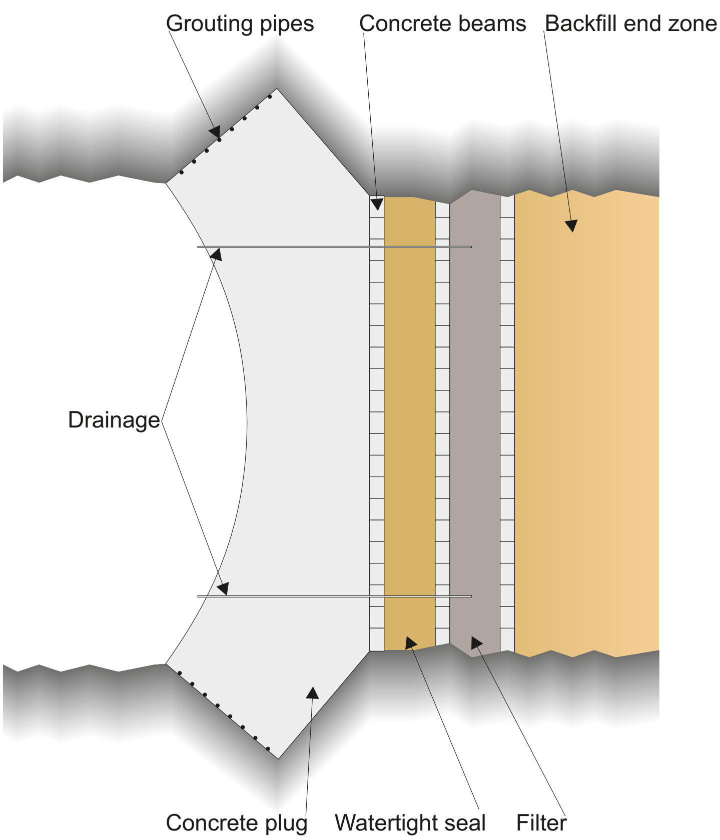
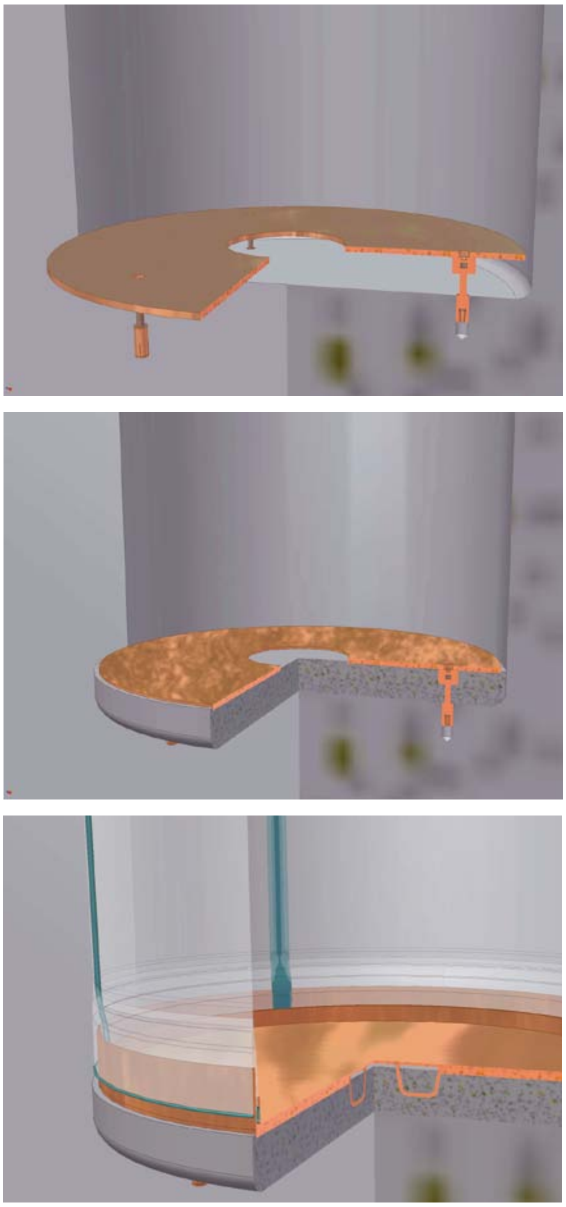


Figure 5-22. *Schematic section of the reference design of the plug.*

Bottom plate

The reference method for drilling deposition holes will not accomplish a flat bottom. In order to achieve a sufficiently flat bottom of the deposition hole, a bottom plate is installed. The reference bottom plate consists of a low pH-cement concrete slab, and a lower and upper copper plate. At installation three bolts are fixed in the rock at the bottom of the deposition hole. The 20 mm thick lower copper plate is placed on top of the bolts. The bolts are then used to adjust the copper plate into a horizontal position. After that the concrete is poured through a hole in the centre of the lower copper plate, forming a 150 mm thick layer. Finally the 10 mm thick upper copper plate is placed in the deposition hole. The bottom plate is illustrated in Figure 5-23. A more detailed description of the bottom plate is given in /Wimelius and Pusch 2008/.



Lower copper plate resting on bolts fixed to the rock.

Thickness	20 mm
Diameter	1,710 mm
Hole diameter	450 mm
Nominal weight	382 kg

Concrete slab poured through the hole in the centre of the lower copper plate.

Thickness	150 mm
Nominal weight	650 kg
Concrete recipe	See Table 5-1

Upper copper plate with fastening devices and a border intended for auxiliary equipment for the installation of the buffer.

Thickness	~10 mm
Diameter	1,710 mm
Nominal weight	246 kg

Figure 5-23. *The bottom plate in the deposition hole* ***Underground openings construction production report****.*

Borehole seals

A number of investigation boreholes, holes drilled both from the surface and from underground openings have to be sealed at the closure of the deep repository. In the layout of the final repository facility the locations of the boreholes are considered, to ensure that boreholes connected to the surface do not intersect underground openings. Furthermore, deposition holes must not be intersected by any investigation boreholes. The geometry of a borehole seal is mainly determined by the dimensions of the drilled holes. The length of surface-based boreholes ranges from a few metres to more than 1,000 metres and the diameter ranges from 56 to 120 mm. The tunnel-based boreholes are expected to have a length of a few hundred metres and a diameter of 56 to 76 mm. The shallowest parts of the boreholes may have larger diameters. Some boreholes may be more or less horizontal.

Highly compacted bentonite will be used where tight seals are needed and cement-stabilised plugs will be cast where the boreholes pass through fracture zones, see Figure 5-24. The borehole plug will consist of cylindrical pre-compacted clay blocks placed in perforated copper tubes.

For the reference design MX-80 bentonite is chosen. To prevent erosion during the installation phase the bentonite is pre-dried to a water content of about 6% and then compacted to a dry density of 1,900 kg/m^3. The bentonite blocks are contained in perforated copper tubes that are jointed as they are inserted into the holes. The copper tubes provide mechanical protection against abrasion in the installation phase. For a borehole with a diameter of 80 mm the perforated copper tube of the reference design has an outer diameter of 76.1 mm and an inner diameter of 72.1 mm. The tubes have a perforation ratio of 50% with 10 mm diameter holes in order to allow the bentonite to swell into the volume between the tube and the rock.

Along sections where the borehole passes water-conducting fracture zones the bentonite could potentially erode. In such positions the holes are therefore filled with silica concrete, which is a permeable and erosion-resistant material, see Figure 5-24. Further details are found in /Pusch and Ramqvist 2007/.

The upper part of boreholes connected to the surface will be sealed with material that can sustain the swelling pressure exerted by the clay part and offer resistance to mechanical impacts like erosion and glaciations.

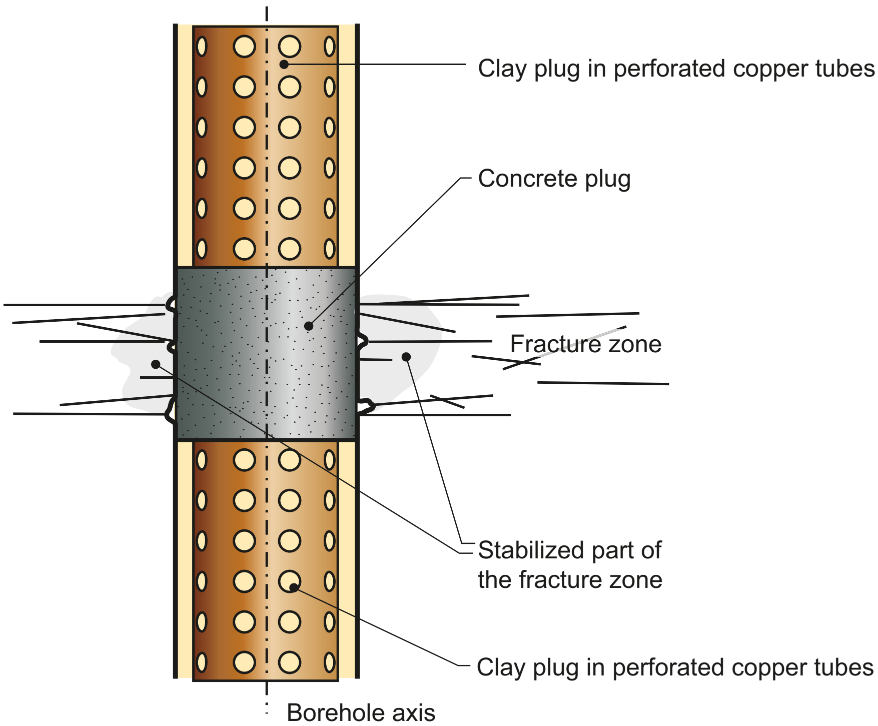


Figure 5-24. *Schematic sketch of the construction of concrete plugs in parts where the borehole passes water-conducting fractures. In the reference design the clay plug is made out of bentonite.*

Closure of main tunnels, ramp and shaft

To conform to the design premises closure of the main tunnels and transport tunnels will be based on the same principle as the backfill in deposition tunnels with blocks and pellets but the material and density will be different. An outline of the reference design for the closure in main tunnels and transport tunnels is given in Section 5.6. In the reference design the ramp and shafts are backfilled with clay up to 200 metres depth, where the top sealing starts. The depth of the top sealing is determined with regard to the presence of water conducting structures and the expected depth of permafrost in Forsmark. Figure 5-25 shows the parts of the ramp and shafts belonging to the top sealing, the deeper parts filled with clay and the use of plugs in ramp and shafts.

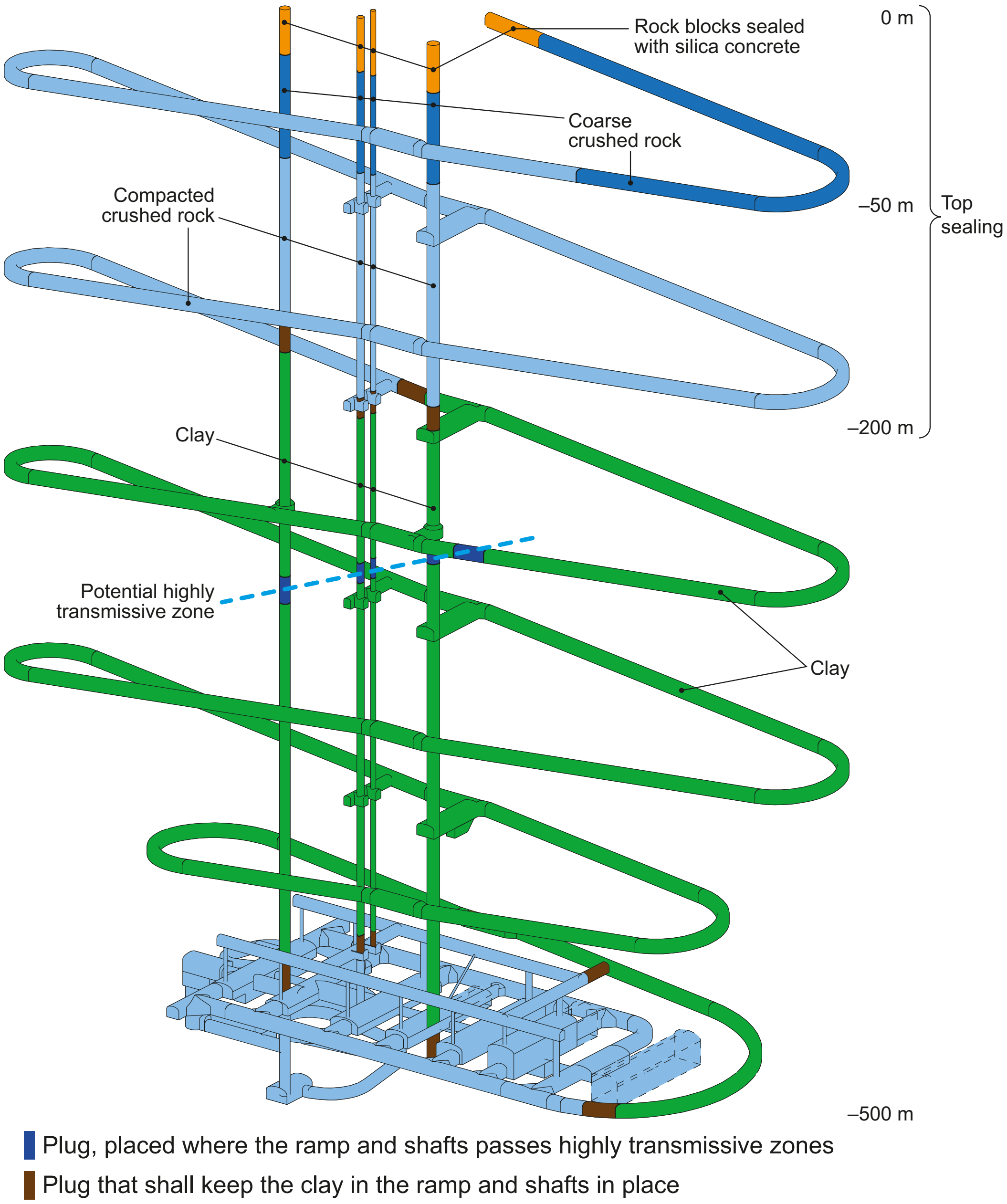


***Figure 5-25.** Top sealing, the parts of the ramp and shafts filled with clay and the placement of plugs in ramp and shafts.*

Top seal

To conform to the requirement that "closure in the upper part of the ramp, shafts and boreholes shall hinder unintentional intrusion into the repository" the ramp and shafts will, from –200 to –50 m depth from the zero level, be filled with crushed rock having a maximum particle size of 200 mm. The rock fill has to be effectively compacted to minimize self-compaction under its own weight and overburden /Pusch 2008/. The uppermost 50 m of the ramp and shafts is planned to be backfilled with very coarse crushed rock that has to be effectively compacted to minimize self-compaction. The shallowest parts of the ramp and shafts will be filled with fairly well fitted blocks of crystalline rock. Good but not perfect fitting of the blocks eliminates arching and the open spaces between the rock blocks are not considered to significantly affect settlement. The reference design for sealing the upper part of the ramp and shafts is indicated in Figure 5-25.

5.7.3 Production procedures

Closure of the repository will, with the exception of some boreholes that have to be closed earlier, not take place until all spent nuclear fuel has been deposited. This means that the closure activities lie well in the future. So far SKB has prioritised the development of the backfill and plug in deposition tunnels. The production of closure for main tunnels and transport tunnels and the ramp and shafts below the level of the top sealing have not yet been developed, but it will most likely resemble production of closure for deposition tunnels. For that reason, this section refers to the **Backfill production report**.

Prior to closure of any underground opening, construction features such as road beds and building components and installations will, as part of the decommissioning, be removed and the underground opening cleaned. Routines for these activities have not yet been specified.

Besides decommissioning, the main stages for closure of the repository are:

- Backfilling of and, if necessary, installation of plugs in any ventilation shaft far away from the central area.
- Backfilling of and, where necessary, installation of plugs in main tunnels and transport tunnels.
- Installation of plugs where the transport tunnels connect to the central area.
- Backfilling of the central area.
- Installation of plugs where the central area connects to shafts and ramp.
- Backfilling of and, where necessary, installation of plugs in the ramp and remaining shafts.
- Installation of the top seal.

Borehole seals are installed at suitable locations before or during the other closure activities.

The production of the bottom plate will be based on available knowledge and conventional technique, SKB foresees no difficulties in manufacturing and installing it in accordance with the reference design.

5.7.4 Initial state

The initial state of the closure is the state when all closure material in a specific underground opening or borehole is installed and the borehole, rock cavity, shaft or tunnel has been closed. Inflow of groundwater to an underground opening or a borehole and its impact on the closure is not accounted for in the initial state, but is assessed as part of SR-Site, see Chapter 10.

The properties of the backfill to be designed to conform to the design premises for long-term safety are:

- Material composition.
- Installed mass (density).
- Installed geometry.

In the **Buffer, backfill and closure process report**, the closure components according to 1–5 in Section 5.7 are characterised by a number of variables. Most of the initial state values for these variables are determined by the design properties. For the clay-filled parts the relationships between the variables and the design parameters are basically the same as for the buffer (see Table 5-14).

Material composition, installed mass, installed geometry

The main components of the installed deposition tunnel plug and the design parameters that shall be inspected in its production are presented in Table 5-22 and the dimensions are shown in Figure 5-26. At this stage of development the presented data should be regarded as illustrative and for some parameters it is not yet possible or meaningful to provide data.

The initial states of the other closure components in the different underground openings and boreholes are presented in Table 5-23. At this stage of development, the closure properties are described as a compilation of reasonable values of some main parameters that can be estimated based on the current results and experience.

The initial state of the bottom plate is that given in Section 5.7.2.

Conformity to design premises

Regarding the backfill in main tunnels, ramp and shaft, the same material and similar production procedures as for the deposition tunnel backfill will be used. Since the maximum allowed hydraulic conductivity in the deposition tunnel backfill is two orders of magnitude lower than that in the main tunnels, ramp and shaft, and since design premises on the deposition tunnel backfill are assumed to be met according to Section 5.6.3, it is assumed that the premises are met also for the main tunnels, ramp and shaft.

Regarding the borehole seals, SKB has, in cooperation with Posiva and as part of SKB's RD&D Programme, developed concepts for sealing long and short boreholes. The concepts are based on the ones tested in Stripa and used in SFR and are judged to work in both steeply and gently plunging boreholes /Pusch and Ramqvist 2007/.

Table 5-22. The main components and design parameters of the installed plug.

Component/Design parameter	Nominal design	Accepted variations
Concrete plug		
Concrete	Low pH concrete-mix B200 (200 kg binder/m^3) according to recipe in the **Backfill production report**.	Will be determined according to standards for conventional concrete
Reinforcement	Quality K500ST[1]	According to standard
	Amount and geometry according to reinforcement drawing[1]	According to reinforcement drawing[1]
Dimensions	According to drawing[2] (see Figure 5-26)	According to drawing[2]
Watertight seal		
Thickness	0.71 m	±10 mm
Material composition	See backfill Section 5.6.2	See backfill Section 5.6.2
Installed dry density	1,500 kg/m^3	To be determined
Filter		
Installed dry density	> 1,900 kg/m^3	To be determined
Thickness	0.7 m	Tolerance of 5%
Beams[3] and other prefabricated components		
Concrete in beams	Low pH concrete mix	Will be determined according to standards for conventional concrete
Reinforcement in beams	Standard steel quality	According to standard
Drainage pipe material	Titanium	According to standard
Drainage pipe dimensions	To be determined	To be determined

1) The reinforcement may be omitted or the amounts determined according to conventional design principles.
2) Detailed drawings will be made according to conventional procedures.
3) Will be designed according to conventional procedures.

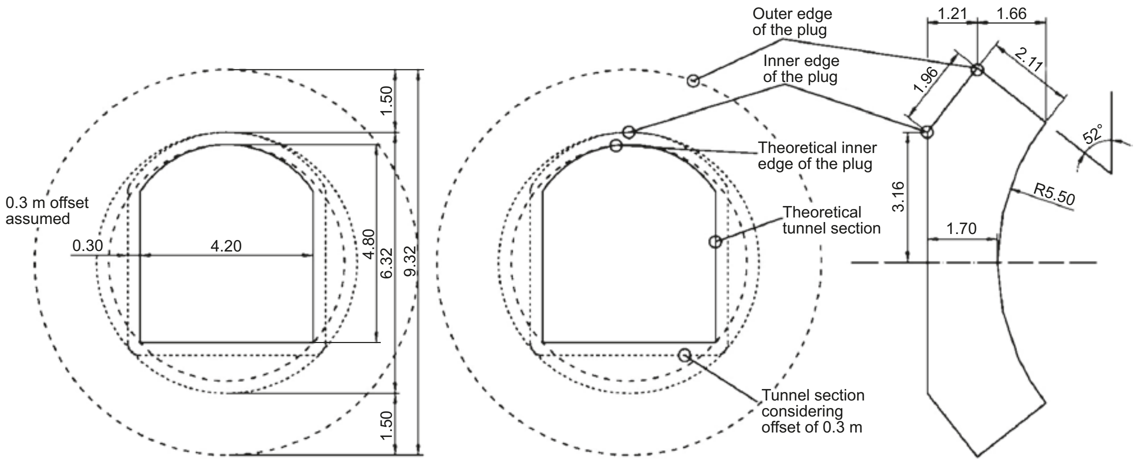


Figure 5-26. *Dimensions of the concrete plug. The rock inside the theoretical inner edge is has not been accounted for in the calculations for how the mechanical load from the plug is transferred to the rock. The offset of 0.3 m between the theoretical tunnel section and the actual tunnel section account for the lookout angle that is a result of the drill and blast excavation method. The excavation damaged zone is deeper in the floor and hence the plug should reach deeper into the rock under the floor.*

Table 5-23. The initial state of the closure in the different underground openings and boreholes.

	Material	Volume (m^3)	Installed dry density (kg/m^3)	Density of solid particles (kg/m^3)	Porosity (%)	Dry weight of installed closure (ton)	Integrated hydraulic conductivity (m/s)
Main tunnels	Clay	390,000	1,460	2,780	47.5	569,400	$< 10^{-8}$
Transport tunnels	Clay	225,000	1,460	2,780	47.5	328,500	$< 10^{-8}$
Central area	Crushed rock	125,000	1,900	2,670	28.8 in the compacted rock fill; 100% in the crown space	237,500	10^{-5}
Ramp (up to level – 200 m)	Clay	115,000	1,460	2,780	47.5	167,900	$< 10^{-8}$
Shafts (up to level – 200 m)	Clay	25,000	1,460	2,780	47.5	36,500	$< 10^{-8}$
Top sealing	Crushed rock	95,000	1,600	2,670	40.1 in the compacted rock fill; 100% in the crown space	152,000	10^{-1}
Boreholes	Clay, copper, concrete	For geometry, see Section 5.7.2	1,630 (clay)	2,780	28.1		$< 10^{-8}$
Plugs	Reinforced concrete						

Examples of results reported in /Pusch and Ramqvist 2007/ are:

- The long-term tests show that a high degree of swelling and homogenisation is obtained after 10–20 days. The measured mean swelling pressure against the rock for the initial dry density 1,905 kg/m^3 of the clay plug core is 2,800 kPa in fresh water and 600 kPa in saline water (Äspö). After saturation and expansion into the slot between the tube and the rock the clay plug will reach a density of 2,025 kg/m^3 which corresponds to a dry density of 1,630 kg/m^3. Measurement of the hydraulic conductivity of the clay paste between tube and rock showed that it was lower than $9 \cdot 10^{-13}$ m/s for saturation and percolation with fresh water and $2 \cdot 10^{-12}$ m/s for saline water.
- Plugging the upper end of deep boreholes can be accomplished by use of copper plugs and CBI silica concrete plugs, both of which can take axial pressures of more than 30 MPa. The shear strength and deformation moduli of concrete plugs of silica concrete can be significantly improved by mixing in centimetre-sized quartzite fragments.
- The function of plugs of the investigated type has been fully demonstrated. For the concrete plugs the recorded strength in full-scale tests has been found to agree well with predicted values derived from laboratory experiments. For clay plugs a very high degree of homogeneity has been documented by an investigation of boreholes that have been plugged for several years. This investigation also demonstrates that the swelling potential and the hydraulic conductivity agree well with predictions.

Tests have also shown that clays with Na as the major adsorbed cation should be used since Ca-saturated clay does not expand readily through the perforation in the copper tube during the saturation phase in either fresh or saline water.

5.8 Monitoring

Repository construction and operation will cause disturbances of the site. Aspects relevant to safety will be handled in the assessment. Monitoring the disturbances will be important for advancing the understanding of the site and the envisaged repository as further outlined in the programme for detailed investigations /SKB 2010b/. There is, however, no legal requirement for monitoring after repository closure. One of the basic requirements is that a final repository should fulfil its function without maintenance and monitoring. The assessment of long-term safety and compliance with applicable regulations is made under these assumptions. Monitoring may still be considered after closure but must then not impair the safe functioning of the repository. SKB's monitoring strategy is evolving but has basically not changed compared to that presented in the SR-Can report /SKB 2006a/. This means that the then cited report, /Bäckblom and Almén 2004/, still provides a valid summary of the current SKB strategy on these matters.

5.8.1 Monitoring for the baseline description

Many of the investigated site parameters like precipitation and groundwater levels show a pattern of more or less pronounced temporal variation. Such variations are for example seasonal fluctuations in temperature and precipitation. Climate change may cause long-term changes or trends in meteorological parameters, which can cause variations in one or several parameters. Furthermore, investigations and underground activities themselves may give rise to changes or variations in values of some parameters.

As set out in the overall SKB strategy for monitoring /Bäckblom and Almén 2004/ and further detailed in the programme for detailed investigations /SKB 2010b/, the purpose of establishing the baseline conditions during the site investigations from surface is to define a reference against which the changes caused by repository development can be recognised and distinguished from natural and man-made temporal and spatial variations in the repository environment.

The description of the baseline conditions is essentially identical to the site descriptive model (see the **Site description Forsmark**) and is based on the data obtained from the site characterisation programme. Part of this characterisation concerns properties that vary with time. Therefore, a monitoring programme, covering both geoscientific and ecological parameters, was initiated during the site investigations, see Section 2.4 of the **Site description Forsmark** and /SKB 2007a/. With a few exceptions, this programme has continued after the completion of the surface based site investigations at Forsmark and will continue once underground excavation work starts.

5.8.2 Monitoring the impact of repository construction

Monitoring the impact of repository construction will be an integral part of the detailed investigation programme /SKB 2010b/ that will be implemented once the underground excavation activities commence. The objective of the monitoring will be to study how repository construction and operation affects the environment. These observations may also provide important data for the hydrogeological and hydrogeochemical modelling and verification of such models. The monitoring is planned to build on the existing monitoring programme /SKB 2007a/. Before implementation of the detailed investigations the adequacy of the existing monitoring programme will be assessed and revised if needed. Particular focus will be on the fact that the programme is intended to operate over a very long time while still being adequate for its main purpose of capturing impacts from construction and operation on the environment. As construction and operation proceeds there will be a need to regularly reassess the selection of monitoring parameters, monitoring objects and measurement frequencies. It should also meet the needs of the environmental control programme and control of repository construction.

5.8.3 Control programme for repository construction and operation

A control program will be developed prior to excavation, with the objective of ensuring that the design premises and other requirements on the construction work and on the operations are fulfilled. The control programme will consider:

- Material deliveries.
- Workmanship.
- Control of the as built and operated facility relative to the design and specification of operational activities.

The control programme with its quality documentation is the basis for assessing whether the construction and operational work conform to the stated design premises and requirements on efficiency and quality. The objectives and contents of the control programme will be defined prior to the underground construction work, but will evolve and be adjusted in response to experience gained.

5.8.4 Monitoring after waste emplacement

Repository closure is a stepwise process from consecutively closing a deposition tunnel to closing one or several deposition areas before the whole repository is closed. Monitoring is planned to continue until all waste has been emplaced and closure of the repository facility is commenced. At closure monitoring systems will be decommissioned successively. At that time it must be considered to what extent the closure process itself needs to be monitored.

Monitoring of the engineered barrier system, i.e. canister, buffer and backfill, is not intended for finally disposed waste as emplacement of instrumentation and the necessary cable leads to sensors is likely to impair the safety functions of the engineered barriers.

As mentioned above there are no legal requirements for monitoring after closure. As stated by /Bäckblom and Almén 2004/ rationales for monitoring of the post-closure phase, such as verification of safeguard requirements, may develop. The extent of the post-closure monitoring programme will essentially be determined by decisions made at, or shortly before, closure and it is appropriate that any decisions on post-closure monitoring are taken by the decision-maker at the time of closure.

However, if monitoring after closure is considered the applicable regulations by SSM should be considered. (SSMFS 2008:21 *8§ The impact on safety of such measures that are adopted to facilitate the monitoring or retrieval of disposed nuclear material or nuclear waste from the repository, or to make access to the repository difficult, shall be analysed and reported to the authority*". Furthermore, the recommendation to this paragraph states: "*The safety report for the facility, in accordance with 9 § should show that these measures either have a minor or negligible impact on repository safety, or that the measures result in an improvement of safety, compared with the situation that would arise if the measures were not adopted*."

Since there are currently no plans for post-closure monitoring, and since such monitoring would not be needed to ensure safety, SR-Site gives no consideration to monitoring after waste emplacement, and it is also assumed that such monitoring, were it to be performed, would not have any detrimental impact on long-term safety.

6 Handling of external conditions

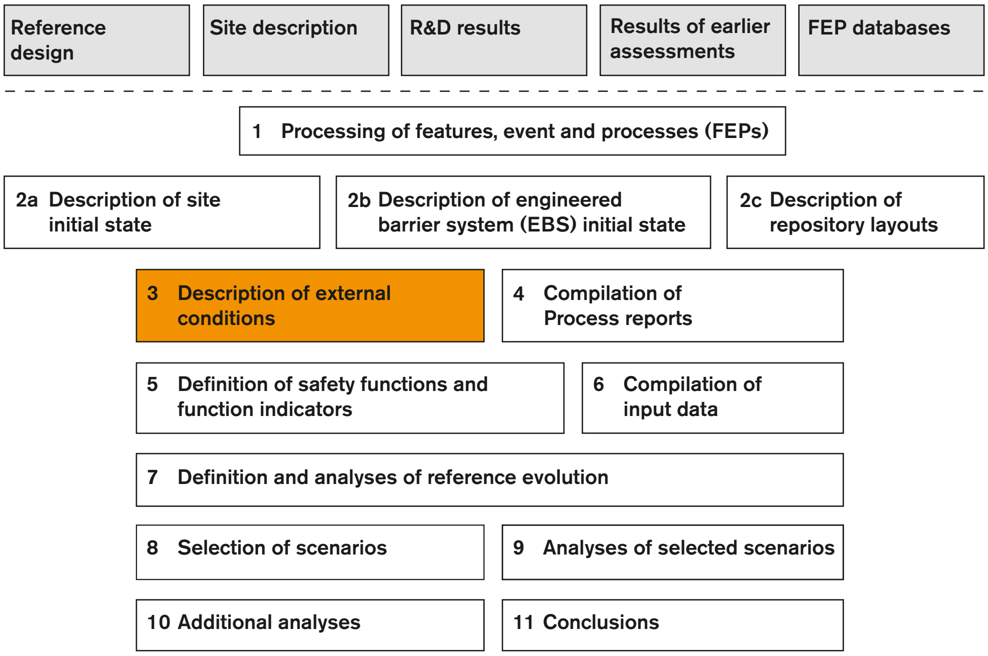


Figure 6-1. *The SR-Site methodology in eleven steps (Section 2.5), with the present step highlighted.*

6.1 Introduction

The external conditions at the repository site are expected to change considerably over the timescale of the safety assessment. External FEPs are one of the main categories in the SR-Site FEP catalogue, see Chapter 3 and the **FEP report**. The external FEPs are further sorted into the following groups.

1. Climate related issues.
2. Large-scale geological processes and effects.
3. Future human actions.
4. Other (only meteorite impact is identified in this group).

The geological processes that can be regarded as external include weathering, erosion, deposition and tectonic uplift (other than induced by glacial loading and unloading, see below) and plate tectonics. In very long time perspectives, millions to hundreds of millions of years, plate tectonic movements, uplift or downwarping, including denudation of the bedrock surface through weathering and erosion, will affect both the geosphere and the Earth climate system.

The tectonic uplift events that formed the present large-scale Fennoscandian topography mainly occurred in the Tertiary (1.8–65 Ma ago), events that raised the Scandinavian mountain range and the South Swedish Dome centred over the Småland county e.g. /Lidmar-Bergström and Näslund 2002/. Although the uplift centres were located west of Forsmark, the site was probably to some extent affected by the uplift. Through increased erosion and weathering, the Tertiary uplift events resulted in a re-exposure of an ancient smooth bedrock surface formed in crystalline bedrock prior to the Cambrian period. Although this smooth bedrock surface, called the Sub-Cambrian peneplain, is somewhat dissected along major fracture zones in the Forsmark region, it has been subject to relatively stable large-scale tectonic conditions during the past 1.8 million years e.g. /Lidmar-Bergström and Näslund 2002, Olvmo 2010/. During this period, isostatic changes due to glacial loading and unloading have dominated the vertical displacement of the site.

/Riis 1996/ suggests that the general rate of bedrock lowering due to erosion and weathering during the Pliocene-Pleistocene (a 5 million year long period ending close to present) has been less than 100 metres per million years for the coastal areas in Sweden. This value describes the general lowering of the bedrock surface, and thus excludes higher erosion rates along for example valleys and bedrock fracture zones. This is in line with results of /Olvmo 2010/ who show that the Forsmark site is located within the intact parts of the Sub-Cambrian peneplain, with extremely low relief, /Olvmo 2010/, resulting in an expected amount of future non-glacial denudation that is very low both in a 100,000 year and 1 million year perspective (up to 5 m per million years for the Forsmark region). Also glacial erosion is estimated to be low in the Forsmark region, around 1–2 m per glacial cycle, see the **Climate report**, Section 3.5, especially locally above the repository since the repository site is not located in a local valley depression or major fracture zone. In summary, the total denudation for the 1 million year assessment period is estimated to be in the order of up to 20 m at Forsmark.

The main effect of this lowering of the ground surface could be that permafrost reaches closer to the repository during periods of periglacial climate conditions. However, given the criteria for freezing of the buffer (Section 8.3.2) and backfill (Section 8.4.4) materials and the estimated depths of the corresponding isotherms in the relevant future climate cases (Sections 10.4.1 and 12.3), the alterations of external conditions caused by weathering and erosion are of minor importance for repository safety within the assessment period. Their impact on the geosphere in the vicinity of the repository and on the current state of the Baltic Shield are reported in the **Climate report** and in the **Geosphere process report** and are not further discussed in this chapter.

Climate changes or climate-related changes, such as the ongoing shore-level displacement, are the most important external factors affecting the repository in a time perspective from tens of years to hundreds of thousands of years. Most of the safety relevant long-term processes occurring in the biosphere and the geosphere are affected by climate and climate-related changes. A safety assessment therefore must address the potential impact of climate change on repository safety. Climate-related issues are further discussed in Section 6.2.

Another main category of external FEPs that may impact the repository is future human actions. These can be divided into actions at or close to the repository site like utilisation of resources from the bedrock and regional or global actions, e.g. those resulting in severe pollution. Future human actions are further discussed in Section 6.3.

The third group of external FEPs in the FEP catalogue contains only the FEP "Meteorite impact". Meteorite impacts have been excluded from further analyses, since the probability is very low that a meteorite, large enough to damage the repository, will actually impact Earth, e.g. on the order of one collision every 500,000 year for objects of roughly 1 km in size /Morbidelli et al. 2002/ and about one collision every 10,000 years for objects causing craters larger than 1 km in diameter /Melosh 1989/. The probability that the hit actually occurs at the repository site is then significantly lower; e.g. an estimated frequency in the order of 10^{-13} per km^2 per year for impacts causing craters larger than 1 km in diameter has been reported by /Hartmann 1965/. Since the depth of a crater is about one third of its diameter /Melosh 1989/, the crater has to be larger than 1 km in order to expose the repository, but the rock would likely be fractured at repository depth due to somewhat smaller impacts. Furthermore, such an impact event would cause substantial damage to the local and regional biosphere, including humans /Collins et al. 2005/, and these direct effects of a meteorite impact are deemed to be much more severe than its possible radiological consequences.

6.2 Climate-related issues

6.2.1 General climate evolution

Natural climate change is caused by factors external to the Earth's climate system and by the complex response of the climate system's components and internal dynamics to those forces. Examples of external natural factors affecting climate in the time perspective of interest for the safety assessment are changes in insolation due to variations of the Earth's orbital parameters, volcanism and solar variability, see the **Climate report**, Sections 2.1 to 2.3. Another factor affecting climate is anthropogenic activities such as burning of fossil fuel, which increases the concentrations of greenhouse gases in the atmosphere. Internal dynamics affecting the climate include those associated with atmospheric and ocean circulation, the waxing and waning of ice sheets and feedback processes such as those relating to temperature – water vapour, ice – albedo, vegetation – albedo and vegetation – precipitation.

The Earth climate system is also closely linked to the carbon cycle, i.e. the continued exchange and reactions of carbon in the terrestrial biosphere, atmosphere, hydrosphere, and sediments, the latter including fossil fuels. There are important feed-back mechanisms in the carbon transfer processes between these carbon reservoirs, many of which have an impact on climate. Global warming could for example suppress terrestrial carbon uptake, which would result in higher carbon dioxide levels in the atmosphere e.g. /Cox et al. 2006/. A description of this topic and related issues is found in /Thorne and Kane 2006/.

Past climate

For the past ~2.5 million years, several cycles of growth and decay of ice sheets have occurred on the Northern Hemisphere and mid-latitudes. Periods during which ice sheets grow and decay are known as glacials. Periods with warm climate when the ice sheets decay to an extent similar to that at the present day are called interglacials. A glacial cycle consists of a glacial and an interglacial. Glacial cycles also include colder and warmer stages termed stadials and interstadials, respectively. Within the glacial phase, the extent of the ice sheets may vary significantly.

Over the last 700,000 years about 100,000 year long glacial-interglacial cycles have dominated climate variability. These cycles consist, generally speaking, of a long period of, in phases, progressively colder conditions followed by a fast transition to a warm interglacial climate. During the glacial periods, ice sheets and glaciers have successively – by repeated advances and decays – grown to a maximum extent, followed by the transition to a warm climate during which the ice sheets rapidly melted away to extents similar to that of the present. At the maximum extent during these cold periods, ice sheets covered about one third of the total land area of Earth (nearly 47 million km^2), compared to at present with ice sheets and glaciers covering about 10% (15 million km^2) of the land surface. One example of proxy data for global climate change during the past 700,000 years is shown in Figure 6-2.

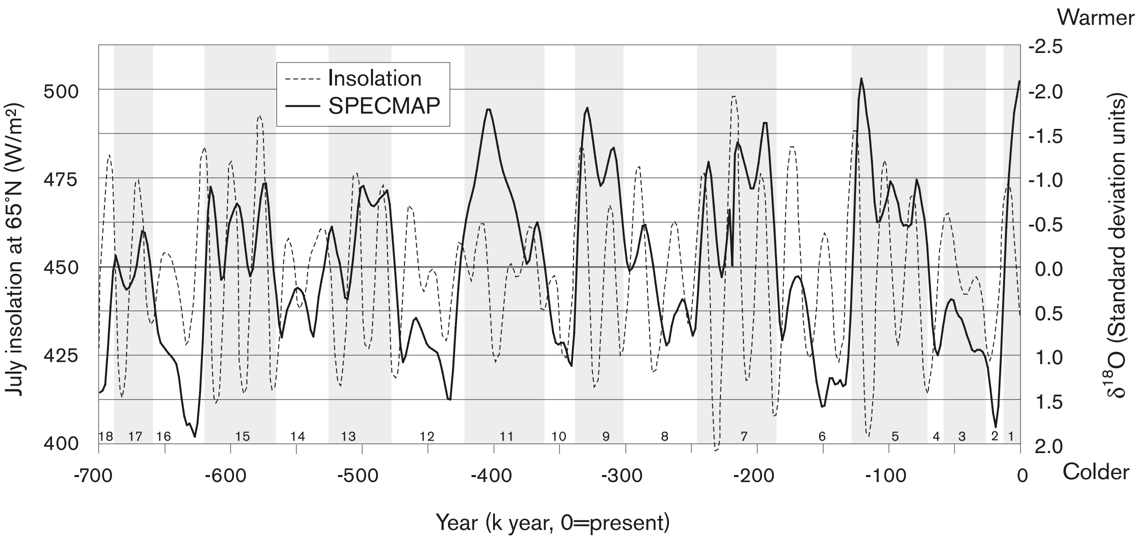


***Figure 6-2.** δ^{18}O variation from five drill cores of deep sea sediments expressed as number of standard deviations from the long-term mean. From /Imbrie et al. 1984/. δ^{18}O variations reflect the temperature of the sea and the volume of water that has been bound in land-based ice sheets and glaciers all over the world. The grey and white fields indicate warm and cold periods.*

Glacials in Sweden

During glacial periods, mountain glaciers in the Scandinavian mountain range expand, eventually forming ice caps that in turn expand into an ice sheet. Due to the cold climate, permafrost typically has developed in the landscape prior to the ice sheet overriding. During the last glacial cycle, the ice sheet grew to its maximum extent c. 20,000 years ago, generally speaking, in a series of ice expansions interrupted by warm interstadials with considerably more restricted ice sheet coverage, see the **Climate report**, Section 4.2.

As the ice sheet grows, the weight of the ice causes an isostatic depression of the Earth's crust. For an overview of the process of glacial isostatic adjustment see the **Climate report**, Section 3.3, /Lund and Näslund 2009/ and /Whitehouse 2009/. Simultaneously, as ice sheets and glaciers expand globally, an eustatic lowering of global sea level occurs as water is moved from the oceans to the land-based ice masses. The net result of the eustatic component (sea-level fluctuations) and isostatic component (vertical changes of the lithosphere) gives a particular development of vertical shore-level displacement. Depending on the relative rate of the eustatic and isostatic processes, the shore-level can either rise or fall, which in turn gives a transgression or regression, respectively. During a deglaciation after a period of major ice sheet coverage, large portions of the coastal regions of Sweden experience a dominant isostatic recovery component with a general regression as a result, interrupted by shorter periods of sea transgressions.

The changes in the position of the shore-line due to the eustatic and isostatic processes will alter the hydrological conditions at the Forsmark site. The ice load and isostatic process will also alter rock stresses. Changes in the stress state introduced by glacial loading and unloading may in places in Sweden lead to bedrock instability /Lund et al. 2009/ and glacially induced faulting /Fälth et al. 2010/.

Further, the basal conditions of the ice sheet are important for the hydrological boundary conditions as well as for effective stresses. Figure 6-3 shows a simplified view of the course of events as an ice sheet grows and decays along a transect from the Norwegian coast towards east.

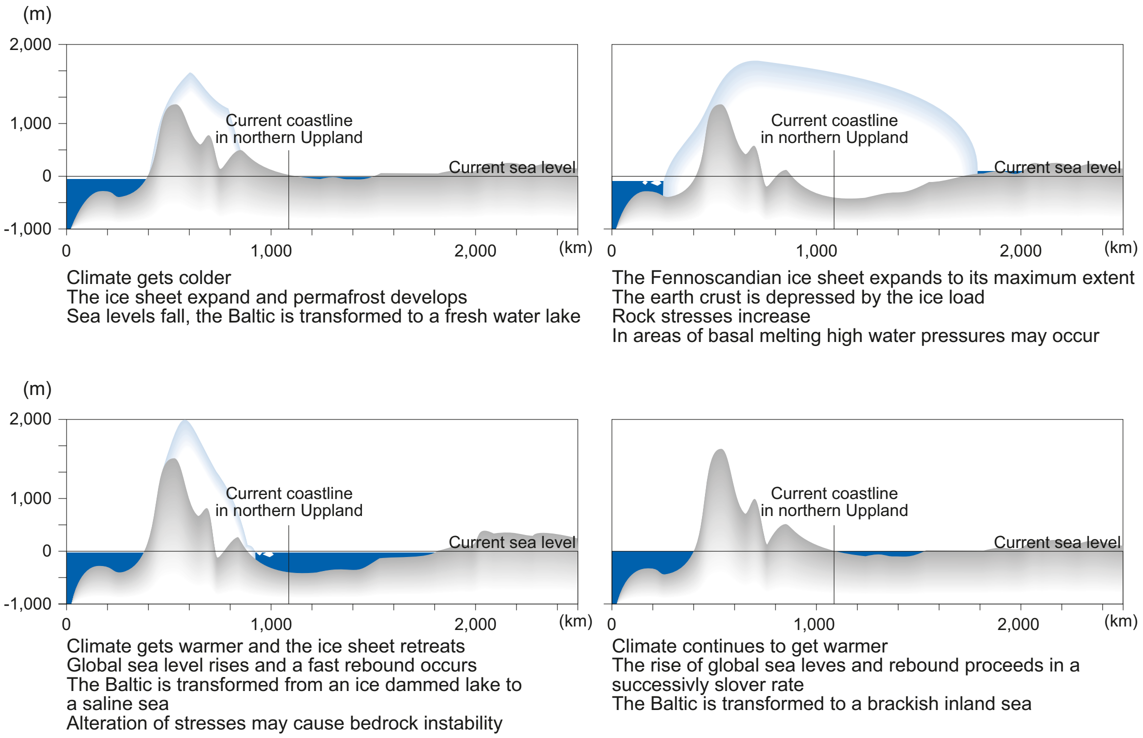


Figure 6-3. *The course of events as an ice sheet expands and decays along a schematic transect from the Norwegian coast towards the east. Note that there is a strong vertical exaggeration of the scale on the y-axis.*

Human-induced climate change

In addition to naturally occurring processes, human emissions of greenhouse gases have been identified by e.g. /IPCC 2007/ as a significant cause of climate change, see the **Climate report**, Sections 5.1 and 5.2. Recent studies utilising both so called Earth Models of Intermediate Complexity and more conventional Atmosphere-Ocean Global Circulation Models have projected a long period of warm climate e.g. /BIOCLIM 2003, 2004, IPCC 2007, Kjellström et al. 2009/. In these model simulations, the emission of greenhouse gases results in a long-term perturbation of the pattern of glacial cycles observed in the past. The perturbation is envisaged to remain until the emitted greenhouse gases have been removed from the surface ocean-atmosphere system and sequestered in the lithosphere. These topics are subject to large uncertainty, for example the knowledge of the carbon cycle is incomplete, the total emissions and their impact on climate are uncertain, and so is the carbon recirculation to the lithosphere /IPCC 2007/. In order to cover such uncertainties, two alternative climate cases are assessed to cover a future climate dominated by global warming.

6.2.2 Impact on repository safety

Climate-related changes such as shore-level migration, development of permafrost and the growth and decay of ice sheets will alter not only surface but also subsurface conditions. Freezing, shore-level displacement and the presence of ice sheets will change permeability, water turnover, groundwater pressures, groundwater flow and composition. The ice load will alter rock stresses and during different phases of a glaciation the principal stresses will change in magnitude. At depths corresponding to the repository depth, the glacial stresses are large enough to change the direction of principal stresses, while this is not the case at large depths. In general, the integrated effects of continuous climatic evolution need to be considered, but there are also a number of more specific phenomena of importance for repository safety that require special attention. Based on the results of earlier assessments, these include:

- The maximum hydrostatic pressure and rock stress occurring at repository depth for glacial conditions.
- The permafrost and freezing depth, affecting freezing of the repository and ground water flow patterns.
- The possible penetration of oxygen-rich groundwater to repository depth during glacial conditions.
- The possible penetration of dilute groundwaters to repository depth during glacial and temperate conditions, potentially causing erosion of buffer and backfill.
- The groundwater salinity occurring at repository depth, both during glacial, periglacial and temperate climate conditions, the latter including also a period of global warming.
- Glacially induced reactivation of faults.
- Factors affecting retardation in the geosphere, such as high groundwater fluxes and mechanical influences on permeability.

6.2.3 Handling the uncertain long-term climatic evolution

The timing and extent of future climate changes are uncertain due to the complexity of the climate system, see the **Climate report**. Additional uncertainty regarding climate evolution is introduced by the uncertain impact and duration of human influence on the climate. It is not possible to predict a single future climate evolution with the confidence needed for assessments of long-term repository safety. However, the *range* within which the climate and climate related processes in Sweden may vary in the future can be estimated with reasonable confidence. Rather than focussing on a most likely climate development, the approach to handling climate uncertainty in the safety assessment is to focus on identifying, describing and analyzing this range including its bounding end-members. Within these limits, characteristic climate conditions can be identified. The conceivable climate conditions can be represented as climate-driven process domains /Boulton et al. 2001/, where such a climate domain is defined as *a climatically determined environment in which a set of characteristic processes of importance for repository safety appear*. The identified domains relevant for the site of the deep repository are:

- A temperate climate domain.
- A periglacial climate domain.
- A glacial climate domain.

The purpose of identifying climate domains is to create a framework for the assessment of climate-related processes of importance for repository safety associated with a particular climatically determined environment. The extent of each climate domain will vary in time and the specific conditions within it will vary both with time and location. The duration of each climate domain depends both on global climate changes and more regional and local factors. The succession of climate domains during a typical glacial cycle will, generally described, follow a cyclic pattern, see Figure 6-4. If a repository for spent nuclear fuel fulfils the safety requirements independent of the prevailing climate domain, and the possible transitions between them, then the uncertainty regarding their extent in time and space is of less importance.

Even if it is not possible to make predictions of the future long-term climate with enough confidence for safety assessment, it is highly likely that the three climate domains will appear repeatedly during the one million year assessment period, i.e. any reasonable future evolution will have to cover them. It is furthermore possible to put bounds on the conditions that plausibly could occur during each of the climate domains. The main scenario of the safety assessment includes a reasonable succession of the identified climate domains.

For compliance purposes, it is particularly important to include sequences covering external conditions yielding, with a high likelihood, the highest risk during the assessment period for a specified scenario of radionuclide release from the repository. Based on results of earlier analyses, the highest risks are likely to occur during temperate periods. Typical situations are i) a terrestrial system that has accumulated radionuclide releases over a long time, possibly in sea sediment prior to its emergence and that is later used for agriculture and ii) a well intruding into the host rock and used for domestic purposes.

It is also important to include the climate domains and sequences that have the greatest impact on repository safety through impairment of barrier safety functions. Phenomena that may impact barrier safety functions are mentioned above; the most severe potential effects are related to the development of permafrost and the advance and decay of ice sheets.

The handling of the climate evolution in SR-Site is further developed in Chapter 11 in conjunction with scenarios and in Section 12.1.3 as an introduction to the analyses of the selected scenarios.

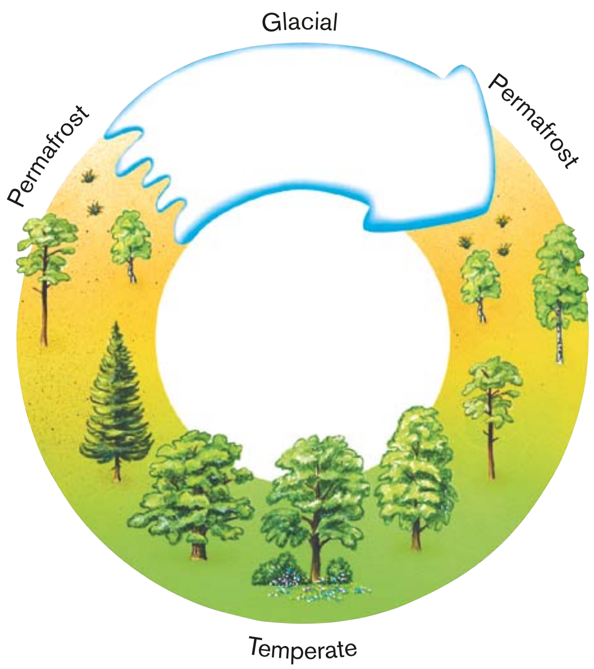


***Figure 6-4.** The climate domains succeed each other in a cyclic pattern. For the analyses in the safety assessment, this picture is refined.*

6.2.4 Documentation

The climate-related conditions and processes identified as relevant for the long-term safety of a KBS-3 repository are identified and described in the **Climate report**, one of the supporting documents for the safety assessment SR-Site, see Section 2.5.12. The purpose of the **Climate report** is to provide a concise description of the climate system and to document the scientific knowledge of the climate-related conditions and processes relevant for the long-term safety of a KBS-3 repository to a level required for an adequate treatment in the safety assessment. The report includes five main parts/chapters:

- Introduction and background.
- The climate system.
- Climate and climate-related issues.
- A reconstruction of last glacial cycle conditions, in turn used for a construction of a future reference glacial cycle for the safety assessment.
- Additional possible climate cases with a potentially larger impact on the long-term safety of a KBS-3 repository than the reference glacial cycle.

"The climate system" chapter includes an overview of the knowledge of the Earth's climate system and a description of the climate conditions that can be expected to occur in Sweden on a 100,000 year time perspective. Based on this summary, climate-related issues relevant to the long-term safety of a KBS-3 repository are identified, such as ice sheets, shore-level and permafrost. These are documented in Chapter 3 "Climate and climate-related issues" to a level required for an adequate treatment in the safety assessment. In Chapter 4, a future 120,000 year long reference glacial cycle, including a characterisation of identified climate-related issues of importance for repository safety, is presented. Finally, relevant complementary climate cases, with a potentially larger impact on repository safety than the reference glacial cycle, are presented in Chapter 5. These cases describe for example situations of expected maximum ice load or maximum permafrost depth. The **Climate report** also includes a description of surface denudation processes (weathering and erosion) as well as a detailed description of the strategy to accommodate long-term climate changes in line with the presentation above.

As further described in the **FEP report**, the content of the **Climate report** has been audited through comparisons with FEP databases compiled in other assessment projects. The **Climate report** follows as far as possible the template for documentation of processes regarded as internal to the repository system, see Section 7.3. However, rather than single processes, a number of more comprehensive climate related *issues* are treated. The issues are i) development of permafrost, ii) ice-sheet dynamics, iii) ice-sheet hydrology and iv) isostatic adjustment and shore-level change, v) surface denudation. Each climate-related issue includes a set of processes together resulting in the behaviour of a system or feature. For instance "ice-sheet dynamics" is the result of several thermal, hydrological and mechanical processes, but, considering the interaction of an ice sheet with its bed, it can be regarded as one entity.

6.3 Future human actions

A great number of external FEPs related to future human actions (FHA) were identified as a result of an audit against the NEA international database conducted for SR-Can and the complementary audit carried out for SR-Site (described in Chapter 3), see further the **FEP report**. These include actions like rock drilling, mining, severe pollution, underground excavations in relation to urbanisation and intentional or inadvertent repository intrusion. In SR-Can, the identified FEPs were briefly audited against the results of the analyses of scenarios based on future human actions carried out in the SR 97 assessment /SKB 1999a/. The majority of the identified FEPs were included in the SR 97 analyses. The latter study was carried out without reference to the NEA database.

The strategy for managing and analysing future human actions in SR-Can /SKB 2006e/ was built on the strategy developed for SR 97 and experience from SR 97, the FEP audit in SR-Can and review of some relevant literature published after SR 97. Some further developments have been made for SR-Site based on experience from SR-Can and more recent literature, whereas the complementary FEP audit carried out in SR-Site did not contribute any new information that needed to be considered. The development of the strategy for handling future human actions in SR-Site is described in the **FHA report** together with the analyses of some representative cases. A summary is provided in Section 14.2.

7 Handling of internal processes

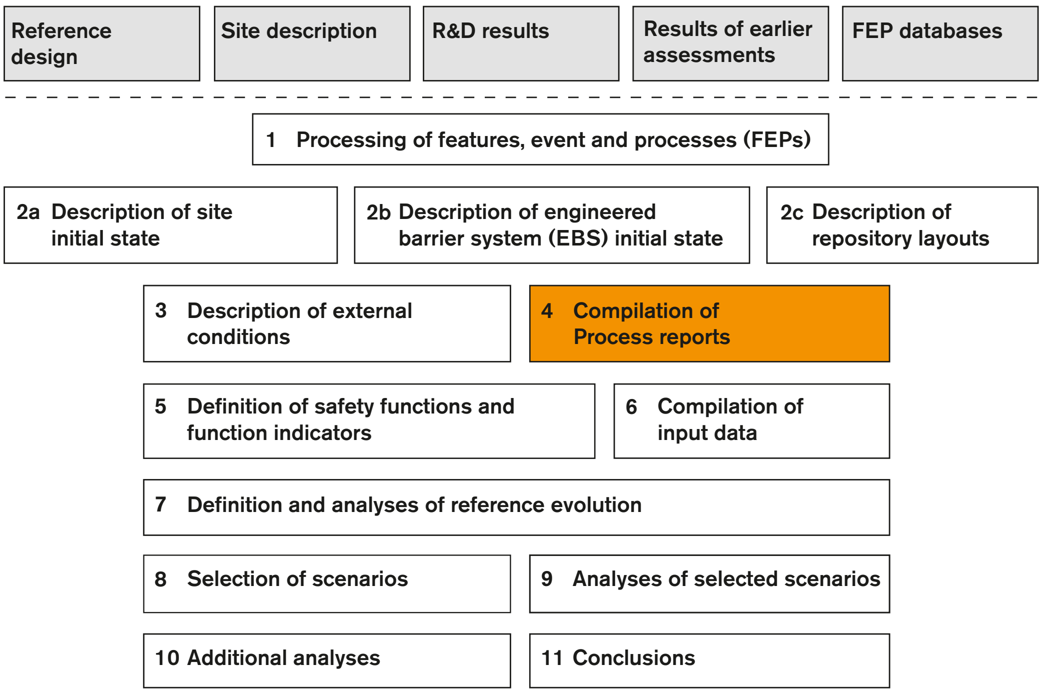


Figure 7-1. *The SR-Site methodology in eleven steps (Section 2.5), with the present step highlighted.*

7.1 Introduction

A thorough understanding and handling of the processes occurring over time in the repository system is a fundamental basis for the safety assessment. The basic sources of information for this are the results of decades of R&D efforts by SKB and other organisations. In a broader sense, these are based on the knowledge accumulated over centuries of scientific and technological development. The R&D efforts have led to the identification and understanding of a number of processes occurring in the engineered barriers and the natural systems relevant to long-term safety. For the purpose of the safety assessment, the relevant process knowledge for the engineered barriers and the host rock is compiled in a number of **Process reports** which also, for each process, contain a prescription for its handling in the safety assessment. Also short-term geosphere processes/alterations due to repository excavation are included.

This chapter describes how processes are documented in the SR-Site **Process reports**, including the principles for their handling in the safety assessment taking into account relevant uncertainties. Formats for graphically illustrating the system of coupled processes are discussed in Section 7.2. The format for process documentation in the SR-Site **Process reports** is described in Section 7.3. Section 7.4 gives an overview of the handling of all processes in SR-Site, based on the material in the **Process reports**.

7.1.1 Identification of processes

The identification of relevant processes has been a continuing effort over many years, based on R&D results, findings in earlier safety assessments etc. In the SR 97 assessment, an identification of the set of processes to be managed in the safety assessment was made /Pers et al. 1999/ and this set was the starting point for process identification in SKB's most recent safety assessment, SR-Can.

As mentioned in Chapter 3, in an audit against the contents of the international FEP database carried out for the SR-Can assessment, a large number of FEPs were mapped to the set of relevant processes in the SKB database leading also to the identification of a few additional processes relevant to the engineered barriers or the geosphere. The updated audit for SR-Site did not lead to the identification of new processes.

In the SR-Can assessment, the division of the system into components was revised and refined, yielding the division given in Section 5.1.2. The deposition tunnel backfill was included as a distinct system component, rather than being described together with the buffer as in SR 97. Also, the components "bottom plate in deposition hole", "plugs", "borehole seals" and "backfill of other repository cavities" were added, but no process reports were developed for them as they are in general not crucially linked to safety. In SR-Site they are included in the **Buffer, backfill and closure process report**.

7.1.2 Biosphere processes

An interaction matrix for the biosphere was developed in the SAFE project /Kautsky 2001/. The result of that work and its further development has been used when setting up the site investigation programme, and it has also provided important input for the development of models for radionuclide transport and exposure to humans. As mentioned in Section 3.3, biosphere processes were not included in the SR 97 Process report and there was not the same basis for updating these descriptions in SR-Can as for the engineered barriers and the geosphere.

For SR-Site, a biosphere process report has been developed /SKB 2010c/. This report contains general descriptions of the processes considered to be of importance for the safety assessment, whereas the site-specific aspects of the processes and how these are handled in the safety assessment are provided in the various ecosystem reports developed for SR-Site /Andersson 2010, Aquilonius 2010, Löfgren 2010/.

7.2 Format for process representations

For the purpose of the safety assessment, the repository system is divided into several system components and each component is characterised by a number of specified time-dependent physical variables, Section 3.3. Within a specific system component, a number of processes act over time to alter the state of the system, i.e. changing the variables. Examples from the buffer are heat transport, water uptake, swelling, chemical decomposition and ion exchange.

The coupling between the processes is expressed by the network of connected processes and variables and the system of coupled processes needs to be managed in the safety assessment. Couplings between system components are, if required, handled via the time-dependent boundary conditions at the component interfaces.

Variables, processes and their dependencies may be graphically represented in different ways. In SR 97, the representation was in the form of Process Diagrams, one for each system component. Figure 7-2 shows the SR 97 process diagram for the buffer. The diagram also shows which variables influence each process as well as the influences a particular process has on the set of variables. Also, interactions across the boundaries of the system component are described. Another example of graphical representation is the Interaction Matrix, e.g. /Skagius et al. 1995/. Both these representations condense a vast amount of information graphically. Both interaction matrices and process diagrams have historically and in SR-Site been used to force the analyst to work in a structured way in identifying relevant processes and barrier properties and their dependencies.

The benefits of the structured treatment of processes and variables in the process diagrams are utilised in the process documentation in the SR-Site **Process reports**. For each process, a table is given describing, for each variable in the system component, if it influences or is influenced by the process in question, see further Section 7.3. For a given process, the table will thus correspond to the arrows for that process in Figure 7-2. The table format also allows comments to be included and thus gives a fuller description than the arrows in the diagram. In conjunction with the table, the influences are documented.

Both interaction matrices and process diagrams do, however, share a difficulty: It is difficult to grasp system evolution by studying these graphical representations. The graphical information in the diagrams or matrices has, therefore, not been utilised directly in the safety assessment, e.g. for illustrating the evolution of the system. The interaction matrices and the process diagrams convey a (true) impression of complexity, but often with no clear guidance as to the relative importance of different traits of system evolution. This is partly related to the lack of distinction between the

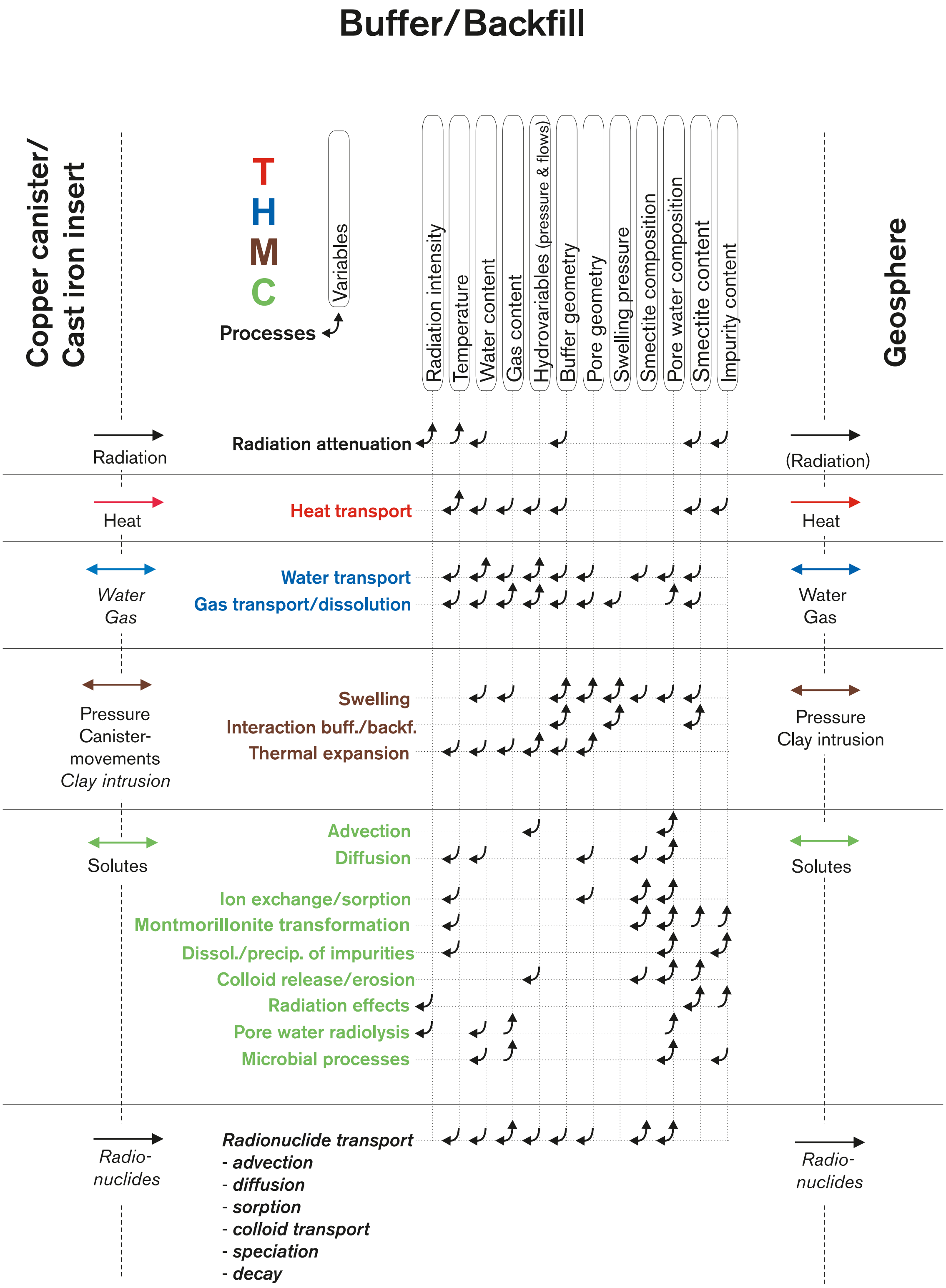


Figure 7-2. *The SR 97 version of the process diagram for the buffer. Thermal, hydraulic, mechanical and chemical processes are listed in the left column, the variables are given in the top row. Influences between variables and processes are shown by arrows in the diagram. Processes and interactions in italics only occur if containment by the copper canister is lost.*

different time frames of repository evolution. Most processes and influences on barrier properties are only relevant in some of the several time frames that need to be considered in the safety assessment. The SR-Site FEP catalogue can generate process diagrams, and, if required, be developed so that interaction matrices can also be generated, but neither of these representations has a central role in the further analyses in SR-Site.

The graphical representation of processes in SR-Site is in the form of tables, where the handling of the processes in different time frames is explained. This mode of presentation is further developed in Section 7.4.

7.3 Format for process documentation

The SR-Site **Process reports** document all processes in the fuel, the canister, the buffer, the backfill and the host rock identified as relevant for long-term safety of a KBS-3 repository as discussed in Section 7.1.1. In addition, processes for the system components bottom plate in deposition holes, tunnel plugs, central area, top seal and borehole seals, which were not included in the process reports in SR-Can, are in SR-Site documented in the **Buffer, backfill and closure process report**. As mentioned in Section 7.1.2, biosphere processes are treated in various biosphere reports and the process descriptions do not follow the format applied in the process reports for the other system components. The main reason for this discrepancy is that the description of processes in the biosphere involves a number of different scientific disciplines, in contrast to the discipline-specific descriptions in the geosphere.

The purpose of the **Process reports** is to document the scientific knowledge of the processes to a level required for an adequate treatment in the safety assessment SR-Site. Therefore, from a scientific point of view, the documentation is not fully comprehensive or highly detailed, since such a treatment is neither necessary for the purposes of the safety assessment nor possible within the scope of an assessment.

The purpose is further to determine an approach to the handling of each process in the safety assessment and to demonstrate how uncertainties are taken care of given the adopted handling.

Generally, all arguments including bases for decisions, and underpinning references are provided in the process description under the appropriate headings. In addition, the expert(s) that have assembled the basic information on each process and the expert(s) that were involved in the decision regarding treatment in the safety assessment are documented in the **Process reports**. All these experts are included in the SR-Site list of experts as required by the SR-Site QA plan, see Section 2.9.4.

All identified processes are documented using a template, where, in essence, all of the headings are the same as those used in the SR-Can versions of the reports. The template is provided below.

Overview/General description

Under this heading, a general description of the knowledge regarding the process is given. For most processes, a basis for this is the contents of the SR-Can version of the process reports, but reviewed and updated as necessary.

Dependencies between process and variables

For each system component, a set of physical variables that defines the state of the system is specified (see Section 3.3). For each process identified according to Section 7.1.1, a table is presented under this heading with documentation of how the process is influenced by the specified set of physical variables and how the process influences the variables. In addition, the handling of each influence in SR-Site is indicated in the table. In all cases where an influence is present, but not handled in the analyses, a justification for the neglect is provided in the table, and/or in the process description. In the latter case, a reference to the text is included in the table. An example of an influence table is provided in Table 7-1.

Table 7-1. Influence table for the process "heat transport" in the geosphere.

Variable	Variable influence on process			Process influence on variable		
	Influence present? (Yes/No Description)	Time period/ Climate domain	Handling of influence (How/If not – Why)	Influence present? (Yes/No Description)	Time period/ Climate domain	Handling of influence (How/If not – Why)
Temperature in bedrock	Yes. Temperature gradients are the driving force for heat transport. Thermal conductivity and heat capacity are temperature dependent.	Excavation/ operation.	Heat transport neglected (see Section 2.1.7 in the **Geosphere process report**).	Yes.	Excavation/ operation	Heat transport neglected (see Section 2.1.7 in the **Geosphere process report**).
		Temperate.	Site-specific temperature and thermal properties. Dependence of thermal properties on T accounted for in dimensioning calculations. Otherwise thermal properties for constant T.		Temperate	Output from calculations.
		Periglacial Glacial.	See Temperate above and the **Climate report**.		Periglacial Glacial	Output from calculations, see also Section 2.2 in the **Geosphere process report** and the **Climate report**
Groundwater flow	Yes.	Excavation/ operation.	Heat transport neglected (see Section 2.1.7 in the **Geosphere process report**).	No. But indirectly through temperature.	–	See Section 3.1 Groundwater flow in the **Geosphere process report**.
		Temperate Periglacial Glacial.	Influence of convection neglected; small contribution compared with thermal conduction.			
Groundwater pressure	Yes.	Excavation/ operation.	Heat transport neglected (see Section 2.1.7 in the **Geosphere process report**).	No. But indirectly through temperature.	–	See Section 3.1 Groundwater flow in the **Geosphere process report**.
		Temperate.	Influence neglected; little significance compared with other influences.			
		Periglacial Glacial.	See Temperate above and the **Climate report**.			
Gas phase flow	Yes.	Excavation/ operation.	Heat transport neglected (see Section 2.1.7 in the **Geosphere process report**).	No. But indirectly through temperature.	–	See Section 3.2 Gas flow/dissolution in the **Geosphere process report**.
		Temperate.	Influence neglected; little significance compared with other influences.			
		Periglacial Glacial.	See Temperate above and the **Climate report**.			
Repository geometry	Yes. Affects heat flux from repository. Canister spacing particularly important in the near field.	Excavation/ operation.	Heat transport neglected (see Section 2.1.7 in the **Geosphere process report**).	No.	–	–
		Temperate	Included in model.			
		Periglacial Glacial.	Included in permafrost model (see the **Climate report**).			
Fracture geometry	Yes.	Excavation/ operation.	Heat transport neglected (see Section 2.1.7 in the **Geosphere process report**).	No. But indirectly through rock stresses and temperature.		See mechanical processes in Chapter 4 in the **Geosphere process report**.
		Temperate.	Influence neglected; little significance compared with other influences.			
		Periglacial Glacial.	Influence neglected; little significance compared with other influences.			

Variable	Variable influence on process			Process influence on variable		
	Influence present? (Yes/No Description)	**Time period/ Climate domain**	**Handling of influence (How/If not – Why)**	**Influence present? (Yes/No Description)**	**Time period/ Climate domain**	**Handling of influence (How/If not – Why)**
Rock stresses	No.	–	–	No. But indirectly through temperature.		See mechanical processes in Chapter 4 in the **Geosphere process report**.
Matrix minerals	Yes. Determines thermal properties.	Excavation/ operation.	Heat transport neglected (see Section 2.1.7 in the **Geosphere process report**).	No.	–	–
		Temperate.	Use of site-specific thermal properties.			
		Periglacial Glacial.	Use of site-specific thermal properties in permafrost model (see the **Climate report**).			
Fracture minerals	Yes. Marginally and locally.	Excavation/ operation.	Heat transport neglected (see Section 2.1.7 in the **Geosphere process report**).	No. But indirectly through temperature and groundwater composition.	–	See chemical processes in Chapter 5 in the **Geosphere process report**.
		Temperate.	Influence neglected; little significance compared with other influences.			
		Periglacial Glacial.	Influence neglected; little significance compared with other influences (see the **Climate report**).			
Groundwater composition	No.	–	–	No. But indirectly through temperature.	–	See chemical processes in Chapter 5 in the **Geosphere process report**.
Gas composition	No.	–	–	No.	–	–
Structural and stray materials	No.	–	–	No.	–	–
Saturation	Yes. Affects scope and extent of convective heat transport.	Excavation/ operation.	Heat transport neglected (see Section 2.1.7 in the **Geosphere Process report**).	No. But, indirectly through temperature.	–	–
		Temperate.	Influence neglected; little significance compared with other influences.			
		Periglacial Glacial.	See Temperate above and the **Climate report**.			

Several reasons for neglect of an influence can be distinguished. These include:

- Little intrinsic significance.
- Little significance compared with other influences.
- Can be subsumed into another influence (without necessarily judging which is the more significant).

Boundary conditions

The boundary conditions for each process are discussed. These refer to the boundaries of the relevant system part. For example, for buffer processes the boundaries are the buffer interfaces with the canister, the walls of the deposition hole, the backfill and the bottom plate in the deposition hole. The processes for which boundary conditions need to be described are, in general, related to transport of material or energy across the boundaries. For example, for chemical processes occurring within a system component, like illitisation in the buffer, the discussion of boundary conditions relates to the boundary conditions of the relevant transport processes occurring in the buffer, i.e. advection and diffusion.

Model studies/experimental studies

Model and experimental studies of the process are summarised. This documentation constitutes the major source of information for many of the processes.

Natural analogues/observations in nature

If relevant, natural analogues and/or observations in nature regarding the process are documented under this heading.

Time perspective

The timescale or timescales on which the process occurs are documented, if such timescales can be defined.

Handling in the safety assessment SR-Site

Under this heading, the handling in the safety assessment SR-Site is described. Typically, the process is either;

- neglected on the basis of the information under the previous headings,
- neglected provided that a particular condition is fulfilled, e.g. that the buffer density is within a specific range, or
- included by means of modelling.

The following aspects are covered, although no prescribed format for the documentation is given.

Time periods: Over what time periods is the process relevant for the system evolution? In e.g. the case of the buffer, relevant time periods might be:

- The resaturation phase extending from the time of deposition until the time when the buffer is fully water saturated.
- The so called thermal phase extending from the time of deposition and throughout the approximately 1,000 year time period of elevated temperature in the buffer. or
- The long-term timescale extending throughout the one million year assessment period and including the varying conditions in the bedrock caused by long-term climate and other environmental variations.

By documenting the relevance of the process for applicable time periods, the process system can be simplified by omitting the process in time periods during which it is not relevant.

Boundary conditions: How are the boundary conditions handled? Are e.g. spatially and temporally varying chemical and hydraulic conditions considered?

Influences and couplings to other processes: The handling of the documented influences is discussed as are couplings to other processes within the system component.

The special cases of a failed canister and of earthquakes altering deposition hole or tunnel geometry: These special cases imply altered conditions that could influence many processes in particular for the fuel, the canister, the buffer and the backfill and they therefore need to be discussed separately. Canister failures and earthquakes of a magnitude that could affect the deposition hole or tunnel geometry are not expected during the several thousands of years after deposition when temperate conditions are likely to prevail, meaning that the special cases are not relevant for many "early" processes.

As a result of the information under this subheading, a mapping of all processes to method of treatment and, in relevant cases, applicable models is produced, see further Section 7.4.

Handling of uncertainties in SR-Site

Given the adopted handling of each process in the safety assessment SR-Site, the handling of different types of uncertainties associated with the process is summarised.

Uncertainties in mechanistic understanding: The uncertainty in the general understanding of the process is discussed based on the available documentation and with the aim of answering the question: Are the basic scientific mechanisms governing the process included and understood to a level necessary for the adopted handling? Alternative models are sometimes used to illustrate this type of uncertainty.

Model simplification uncertainties: In most cases, the quantitative representation of a process contains simplifications. These may result in a significant source of uncertainty in the description of the system evolution. Alternative models or alternative approaches to simplification for a particular conceptual model are sometimes used to illustrate this type of uncertainty.

Input data uncertainties: The set of input data necessary to quantify the process for the suggested handling is documented. The further treatment of important input data and input data uncertainties is described in the **Data Report**, to which reference is made if relevant.

Adequacy of references supporting the handling in SR-Site

Under this heading, statements are provided concerning the adequacy of the references in a quality assurance perspective. These statements are restricted to the references supporting the selected handling and they are evaluated in the factual review of the **Process reports**, together with the arguments and justifications for the selected handling that are provided in the preceding subsections of the process description (see also Section 2.9.5).

References

A list of references used in the process documentation is given at the end of the reports.

7.4 Process mapping/process tables

To summarise the handling of processes in the safety assessment, a table showing the handling of each process has been produced, based on the handling documented in the **Process reports**. The description is broken down in different time frames where relevant. One table per system component is provided. In the table, the process is either "mapped" to a model by which it is quantified or associated with a brief verbal description of how it is handled. The SR-Site process tables for the fuel, the canister, the buffer, the deposition tunnel backfill and the geosphere are presented in Sections 7.4.1 to 7.4.5.

In Section 7.5, two flow charts demonstrate how the different modelling activities are connected.

7.4.1 Fuel and canister interior

Table 7-2. Process table for the fuel and canister interior describing how fuel processes and processes occurring in the canister interior are handled for intact canisters and in the special cases of failed canisters (italicised text). Green fields denote processes that are neglected or irrelevant for the period of interest. Red fields denote processes that are quantified by modelling in the safety assessment. Orange fields denote processes that are neglected subject to a specified condition. Motives for the adopted handling are given in the Fuel and canister process report. An overview of all modelling activities, with references to sections in this main report where the results are discussed, is given in Section 7.5. Much of this information is also given in the last column in the table.

	Intact canister	FEP chart item intact can (see Section 8.5)	Failed canister	Reference to sections in this report (if relevant) and other notes
F1. Radioactive decay	Thermal model.	Decay, heat generation.	COMP23.	In thermal calculation, see Section 10.3.4, in nuclide transport calculations Section 13.4.1.
F2. Radiation attenuation/heat generation	Thermal model.	Decay, heat generation.	Neglected when releases occur after period of elevated temperatures.	In thermal calculation, see Section 10.3.4.
F3. *Induced fission (criticality)*	Neglected since there will be insufficient amounts of moderator inside the canister prior to failure.	–	Neglected since the probability is negligibly small if credit is taken for the burn-up of the fuel.	See further Section 13.3.
F4. Heat transport	Thermal model.	Heat conduction.	Neglected when releases occur after period of elevated temperatures.	In thermal calculation, see Section 10.3.4.
F5. *Water and gas transport in canister cavity, boiling/condensation*	Not relevant.	–	Description in the **Fuel and canister process report**, Section 2.3.1, integrated with other relevant processes yielding simplified, pessimistic assumptions on retardation in failed canister depending on failure mode.	See the **Radionuclide transport report**.
F6. Mechanical cladding failure	Not relevant.	–	Pessimistic assumption.	
F7. Structural evolution of fuel matrix	Not relevant.	–	Negligible for the fuel types and burnup relevant for SR-Site.	
F8. *Advection and diffusion*	Not relevant.	–	Description in the **Fuel and canister process report**, Section 2.3.1, integrated with other relevant processes yielding simplified, pessimistic assumptions on retardation in failed canister depending on failure mode.	Refers to diffusion and advection in the canister interior, see Chapter 13. See also process F17.
F9. Residual gas radiolysis/acid formation	Neglected since negligible amounts of corrodants are produced.	–	Not relevant.	See the **Fuel and canister process report**, Section 2.5.2.

	Intact canister	FEP chart item intact can (see Section 8.5)	Failed canister	Reference to sections in this report (if relevant) and other notes
F10. *Water radiolysis*	Neglected.	–	Neglected except for fuel dissolution, see that process.	Initial water consumed by nitric acid formation or cast iron corrosion.
F11. *Metal corrosion*	Not relevant.	–	Modelled in COMP23.	See Section 13.4.1 and 13.5.2, subheading "Release of activation products".
F12. *Fuel dissolution*	Not relevant.	–	Modelled as constant, pessimistic dissolution rate in COMP23.	Chapter 13.
F13. *Dissolution of gap inventory*	Not relevant.	–	Pessimistic, instantaneous.	See Section 13.4.1.
F14. *Speciation of radionuclides, colloid formation*	Not relevant.	–	COMP23.	Precipitation/dissolution handled by COMP23, concentration limits provided in the **Data report**. See Chapter 13.
F15. Helium production	Neglected since the amount of helium produced will not increase the pressure inside the canister enough to affect its mechanical stability.	–	Not relevant.	See the **Fuel and canister process report**, Section 2.5.8.
F16. Chemical alteration of the fuel matrix	Not relevant.		Neglected since it is not deemed to increase the dissolution rate of the fuel.	
F17. *Radionuclide transport*	Not relevant.		COMP23.	Canister interior treated as mixed tank in all transport calculations reported in Chapter 13.

7.4.2 Canister

Table 7-3. Process table for the canister describing how canister processes are handled for intact canisters and in the special case of failed canisters. Green fields denote processes that are neglected or irrelevant for the period of interest. Red fields denote processes that are quantified by modelling in the safety assessment. Orange fields denote processes that are neglected subject to a specified condition. Motives for the adopted handling are given in the Fuel and canister process report. An overview of all modelling activities, with references to sections in this main report where the results are discussed, is given in Section 7.5. Much of this information is also given in the last column in the table.

	Intact canister	FEP chart item intact can (see Section 8.5)	Failed canister	Reference to sections in this report (if relevant) and other notes
C1. Radiation attenuation/heat generation	Radiation attenuation: Initial radiation levels given in the **Canister production report**, referring to calculations in the **Spent fuel report**. Heat generation: Included in integrated modelling of thermal evolution; thermal model.	Decay, heat generation.	Neglected when releases occur after period of elevated temperatures.	See Sections 5.3.4 (initial radiation levels) and 10.3.4 (thermal evolution).
C2. Heat transport	Included in integrated modelling of thermal evolution; thermal model.	Heat conduction.	Neglected when releases occur after period of elevated temperatures.	See Section 10.3.4.
C3. Deformation of cast iron insert	Isostatic and shear loads modelled in design analysis of canister, see Section 5.4.3. (Uneven loads from uneven swelling and lack of straightness of the deposition hole included.) Considered as cause for failure when isostatic or shear loads on canister exceed design premises loads.	Isostatic load. Rock shear.	Description in the **Fuel and canister process report**, Section 3.4.2, integrated with other relevant processes yielding simplified, pessimistic assumptions on retardation in failed canister depending on failure mode.	See Sections 10.3.13 (temperate period), 10.4.5 (large earthquakes) and 10.4.9 (isostatic pressure for glacial load).
C4. Deformation of copper canister from external pressure	Isostatic and shear loads modelled in design analysis of canister, see Section 5.4.3. (Uneven loads from uneven swelling and lack of straightness of the deposition hole as well as creep in copper included.) Considered as cause for failure when isostatic or shear loads on canister exceed design premises loads.		Neglected.	See in particular Section 10.4.5 (large earthquakes).
C5. Thermal expansion (both cast iron insert and copper canister)	Neglected since the thermal expansion will cause negligible strains in the materials.	–	Neglected.	
C6. *Copper deformation from internal corrosion products*	Not relevant.	–	Description in the **Fuel and canister process report**, Section 2.3.1, integrated with other relevant processes yielding simplified, pessimistic assumptions on retardation in failed canister depending on failure mode.	See the **Radionuclide transport report**.
C7. Radiation effects	Neglected based on the specified limit on Cu content for the insert (Cu impurities promote radiation damage). No radiation effect on the copper shell.	–	Neglected.	

	Intact canister	FEP chart item intact can (see Section 8.5)	Failed canister	Reference to sections in this report (if relevant) and other notes
C8. *Corrosion of cast iron insert*	Not relevant.	–	Description in the **Fuel and canister process report**, Section 2.3.1, integrated with other relevant processes yielding simplified, pessimistic assumptions on retardation in failed canister depending on failure mode.	See the **Radionuclide transport report**.
C9. *Galvanic corrosion*	Not relevant.	–	Neglected, as influence of galvanic corrosion under oxygen-free, reducing conditions lies within the margins of error for the corrosion rate of the iron insert.	See the **Radionuclide transport report**.
C10. Stress corrosion cracking of cast iron insert	Neglected since stress corrosion cracking is considered unlikely and even if it occurred it would have no consequences for stability of the insert.	–	Neglected.	
C11. Corrosion of copper canister	Generally, corrosion is modelled based on mass balance and transport capacity considerations whereas reaction rates are disregarded. Sulphide in buffer and backfill modelled. Microbially generated sulphide in buffer pessimistically bounded by supply of nutrients for microbes. Initial oxygen in buffer: Pessimistically assumed that all oxygen corrodes copper neglecting oxygen consumption by buffer pyrite and rock. Initial oxygen in tunnel backfill: Consider consumption by rock and microbes. Potentially intruding oxygen: Integrated handling of rock, backfill and buffer conditions. Pitting (oxygen corrosion): Described as uneven general corrosion. Corrosion due to radiation: Negligible corrosion depths. Chloride assisted corrosion: Neglected if pH > 4 and $[Cl^-]$ < 2 M. Corrosion effects on cold worked material neglected due to small consequences, if any. Corrosion by water modelled as “what-if-case”.	Corrosion.	Not relevant for failed canister.	See Sections 10.3.13 and 10.4.9.
C12. Stress corrosion cracking, copper canister	Neglected due to the combined effect of very low (if any) concentrations of SCC promoting agents and the insufficient availability of oxidants.	–	Neglected.	
C13. Earth currents – stray current corrosion	Neglected due to no increase in corrosion from external electrical field.	–	Neglected.	
C14. Deposition of salts on canister surface	Neglected due to small consequences (and only relevant during bentonite saturation phase).	–	Neglected.	
C15. Radionuclide transport	Not relevant.	–	COMP23.	See Chapter 13.

7.4.3 Buffer

Table 7-4. Process table for the buffer describing how buffer processes are handled in different time frames and for the special case of an earthquake. Green fields denote processes that are neglected or not relevant for the period of interest. Red fields denote processes that are quantified by modelling in the safety assessment. Orange fields denote processes that are neglected subject to a specified condition. Motives for the adopted handling are given in the Buffer, backfill and closure process report. An overview of all modelling activities, with references to sections in this main report where the results are discussed, is given in Section 7.5. Much of this information is also given in the last column in the table.

	Resaturation/ "thermal" period	Long-term after saturation and "thermal" period	Earthquakes	FEP chart item (see Section 8.5)	Reference to sections in this report (if relevant) and other notes.
Intact canister					
Bu1. Radiation attenuation/heat generation	Neglected since dose rate is too low to be of importance for the buffer.	Neglected since dose rate is too low to be of importance for the buffer.	Not relevant.	–	
Bu2. Heat transport	Thermal model	Thermal model.	Not relevant.	Heat conduction.	See Section 10.3.4.
Bu3. Freezing	Neglected, since this requires permafrost conditions.	Neglected if buffer temperature > −4°C. Otherwise bounding consequence calculation.	Not relevant.	Freezing, expansion.	Repository temperature in long term obtained from permafrost depth modelling, see Section 10.4.3.
Bu4. Water uptake and transport for unsaturated conditions	Buffer and backfill THM model.	Not relevant by definition.	Not relevant.	Saturation.	See Section 10.3.8.
Bu5. Water transport for saturated conditions	Neglected under unsaturated conditions, For saturated conditions the treatment is the same as for "Long-term".	Neglected if hydraulic conductivity < 10^{-12} m/s since diffusion would then dominate	See process Bu9.	Advection.	Regarding consequences of buffer colloid release, see Section 10.4.9, subheading "Canister corrosion for a partially eroded buffer".
Bu6. Gas transport/dissolution	Through dissolution.	(Through dissolution) No gas phase is assumed to be present.	(Through dissolution) No gas phase is assumed to be present.	–	
Bu7. Piping/erosion	Quantitative estimate with an empirical model.	Not relevant, see also Bu18.	Not relevant.	Piping/erosion.	See Section 10.2.4.

	Resaturation/ "thermal" period	Long-term after saturation and "thermal" period	Earthquakes	FEP chart item (see Section 8.5)	Reference to sections in this report (if relevant) and other notes.
Bu8. Swelling/Mass redistribution	Buffer and backfill THM modelling including interaction buffer/backfill and thermal expansion.	Integrated evaluation of erosion, convergence, corrosion products, creep, swelling pressure changes due to ion exchange and salinity, canister sinking.	Part of integrated assessment of buffer/canister/rock.	Swelling.	Initial saturation/swelling: See Section 10.3.9. Long-term temperate conditions: Section 10.3.12. Effects on swelling pressure from salinity are discussed in Sections 10.3.9, 10.4.8 and 12.2.2. The effect of ion-exchange on swelling pressure is discussed in Section 12.2.2. Earthquakes: See Section 10.4.5. Consequences of buffer colloid release (erosion) discussed in Section 10.4.9, subheading "Canister corrosion for a partially eroded buffer" and covered by stylised case reported in Section 10.3.9. Canister sinking is discussed in Section 10.4.8 and 12.2.4.
Bu9. Liquefaction	Not relevant.	Neglected since liquefaction from a short pulse cannot occur in a high density bentonite, due to high effective stresses.	Neglected since liquefaction from a short pulse cannot occur in a high density bentonite, due to high effective stresses.		See Section 10.4.8, subheading "Liquefaction".
Bu10. Advective transport of species	Simplified assumptions of mass transport of dissolved species during saturation.	Neglected if hydraulic conductivity < 10^{-12} m/s.	See process Bu9.	Advection.	See process Bu5.
Bu11. Diffusive transport of species	Chemistry model (thermal, saturated phase; unsaturated phase disregarded).	Chemistry model.	Not specifically treated.	Diffusion.	Thermal phase: Section 10.3.10 Long-term temperate: Section 10.3.12 Glacial cycle: Section 10.4.8, subheading "Chemical evolution of buffer and backfill for altered groundwater compositions".

	Resaturation/ "thermal" period	Long-term after saturation and "thermal" period	Earthquakes	FEP chart item (see Section 8.5)	Reference to sections in this report (if relevant) and other notes.
Bu12. Sorption (including ion-exchange)	Chemistry model (thermal, saturated phase; unsaturated phase disregarded).	Chemistry model.	Not specifically treated.	Reactions in porewater and clay, sorption.	See process Bu11.
Bu13. Alterations of impurities	Chemistry model (thermal, saturated phase; unsaturated phase disregarded).	Chemistry model.	Not specifically treated.	Reactions in porewater and clay, sorption.	See process Bu11.
Bu14. Aqueous speciation and reactions	Chemistry model (thermal, saturated phase; unsaturated phase disregarded).	Chemistry model.	Not specifically treated.	Reactions in porewater and clay, sorption.	See process Bu11.
Bu15. Osmosis	Evaluation through comparison with empirical data.	Evaluation through comparison with empirical data.	Not specifically treated.	Osmosis.	Initial saturation/swelling: See Section 10.3.9. Long-term temperate: Section 10.3.12. Glacial cycle: Section 10.4.8, subheading "Effects of saline water on buffer and backfill".
Bu16. Montmorillonite transformation	Model calculations (thermal, saturated phase; unsaturated phase disregarded).	Estimate based on evidence from nature.	Part of integrated assessment of buffer/canister/rock.	Reactions in porewater and clay, sorption.	Thermal phase: See Section 10.3.10, subheading "Mineral transformation", covers also long-term temperate phase.
Bu17. Iron-bentonite interaction	Neglected since no iron will be in contact with the bentonite.	Only considered for failed canister. Possible loss of buffer efficiency.	Only considered for failed canister. Possible loss of buffer efficiency.		
Bu18. Montmorillonite colloid release	Neglected if total cation charge is > 4 mM. Otherwise modelled.	Neglected if total cation charge is > 4 mM. Otherwise modelled.	Not specifically treated.	Colloid release.	Long-term temperate: Section 10.3.12 Glacial cycle: Section 10.4.8, subheading "Colloid release from buffer and backfill".
Bu19. Radiation-induced transformations	Neglected since dose rate outside canister is too low to have any effect.	Neglected since dose rate outside canister is too low to have any effect.	Neglected since dose rate outside canister is too low to have any effect.	–	
Bu20. Radiolysis of porewater	Neglected since dose rate outside canister is too low to have any effect.	Neglected since dose rate outside canister is too low to have any effect.	Neglected since dose rate outside canister is too low to have any effect.	–	

	Resaturation/ "thermal" period	Long-term after saturation and "thermal" period	Earthquakes	FEP chart item (see Section 8.5)	Reference to sections in this report (if relevant) and other notes.
Bu21. Microbial processes	Neglected under unsaturated conditions, since the extent of aqueous reactions is limited. For saturated conditions the treatment is the same as for "Long-term".	Quantitative estimate of sulphate reduction, limited by supply of microbe nutrients in groundwater.	Not specifically treated.	Corrosion.	Sulphate reduction included in corrosion calculation, Section 10.4.9.
Bu22. Cementation	Discussed together with Process Bu16 "Montmorillonite transformation".	Discussed together with Process Bu16 "Montmorillonite transformation".	Part of integrated assessment of buffer/canister/rock.		
Failed canister					
Bu6. Failed canister. Gas transport/dissolution	*(no failures are expected this period)*	Quantitative estimate based on empirical data.	Quantitative estimate based on empirical data.	–	See Section 13.8.
Bu19. Failed canister. Radiation-induced transformations	Neglected since dose rate outside canister is too low to have any effect.	The effect of α-radiation from nuclides from a failed canister is estimated.	The effect of α-radiation from nuclides from a failed canister is estimated.	–	The effect of α-radiation from nuclides is estimated in the **Buffer, backfill and closure process report**, Section 3.5.12. There, it is concluded that the consequences can be neglected.
Bu23. Colloid transport	Neglected if density at saturation > 1,650 kg/m³, otherwise bounding calculation *(no failures are expected this period).*	Neglected if density at saturation > 1,650 kg/m³, otherwise bounding calculation.	Neglected if density at saturation > 1,650 kg/m³, otherwise bounding calculation.	–	See Section 13.5.3.
Bu24. Speciation of radionuclides	*(no failures are expected this period)*	Assumptions based on empirical data	Assumptions based on empirical data.	–	See Chapter 13.
Bu25. Transport of radionuclides in water phase	*(no failures are expected this period)*	COMP23.	COMP23. Reduced diffusion path.	–	See Chapter 13. Earthquake: See Section 13.6.
Bu26. Transport of radionuclides by a gas phase	*(no failures are expected this period)*	Quantitative estimate.	Quantitative estimate.	–	See Section 13.8.

7.4.4 Backfill in deposition tunnels

Table 7-5. Process table for the backfill describing how backfill processes are handled in different time frames. Green fields denote processes that are neglected or not relevant for the period of interest. Red fields denote processes that are quantified by modelling in the safety assessment. Orange fields denote processes that are neglected subject to a specified condition. Motives for the adopted handling are given in the Buffer, backfill and closure process report. An overview of all modelling activities, with references to sections in this main report where the results are discussed, is given in Section 7.5.

	Resaturation/"thermal" period	Long-term after saturation and "thermal" period	FEP chart item (see Section 8.5)	Reference to Sections in this report (if relevant) and other notes
Intact canister				
BfT1. Heat transport	Simplified assumption.	Simplified assumption.	–	
BfT2. Freezing	Neglected, since this requires permafrost conditions.	Neglected if backfill temperature > –2°C. Otherwise discussed.	Not included since freezing in backfill relates to retardation.	Less severe consequences than for buffer.
BfT3. Water uptake and transport for unsaturated conditions	Buffer and backfill THM model.	Not relevant by definition.	Saturation.	The pellets are included in the model.
BfT4. Water transport for saturated conditions	Neglected under unsaturated conditions, For saturated conditions the treatment is the same as for "Long-term".	Included in geosphere modelling.	Advection.	Evaluate effects on conductivity of chemical evolution and mass redistribution/loss and of possible changes of hydraulic gradients for permafrost and glaciation.
BfT5. Gas transport/dissolution	Buffer and backfill THM model.	(Through dissolution).		The presence of a trapped gas phase is considered in the modelling of the saturation of the backfill (not the case for the buffer).
BfT6. Piping/erosion	Quantitative estimate with an empirical model.	Not relevant, see also BfT16.	Piping/erosion.	See also water transport for saturated conditions.
BfT7. Swelling/Mass redistribution	Buffer and backfill THM modelling including interaction buffer/backfill and homogenisation in tunnel.	Integrated evaluation of erosion, convergence, creep, swelling pressure changes due to ion exchange and salinity and transformation.	Swelling (buffer).	Deviations in amount of buffer and backfill initially deposited and buffer saturating before tunnel backfill is discussed in Section 10.3.9. The effect of salinity is discussed in Section 10.3.9 and 10.4.8. The effect of erosion is discussed in Section 10.4.8.
BfT8. Liquefaction	Not relevant.	Not relevant.	–	Less severe consequences than for buffer. Discussed in Section 10.4.8.

	Resaturation/"thermal" period	Long-term after saturation and "thermal" period	FEP chart item (see Section 8.5)	Reference to Sections in this report (if relevant) and other notes
BfT9. Advective transport of species	Simplified assumptions of mass transport of dissolved species during saturation.	Included in geosphere modelling. Cases without the backfill path are considered	Advection.	See "Water transport for saturated conditions".
BfT10. Diffusive transport of species	The early stage is not studied specifically, since the conditions in the backfill will be comparable to those for the long-term evolution.	Chemistry model.	Diffusion.	Thermal phase: Section 10.3.10 Long-term temperate: Section 10.3.12 Glacial cycle: Section 10.4.8, subheading "Chemical evolution of buffer and backfill for altered groundwater compositions"
BfT11. Sorption (including ion-exchange)	The early stage is not studied specifically, since the conditions in the backfill will be comparable to those for the long-term evolution.	Chemistry model.	Reactions in porewater and clay, sorption.	See process BfT10. See also osmosis.
BfT12. Alterations of impurities	The effect on inorganic reduction of oxygen is modelled.	Chemistry model.	Reactions in porewater and clay, sorption.	See process BfT10.
BfT13. Aqueous speciation and reactions	The early stage is not studied specifically, since the conditions in the backfill will be comparable to those for the long-term evolution.	Chemistry model.	Reactions in porewater and clay, sorption.	See process BfT10.
BfT14. Osmosis	Hydraulic conductivity in THM model chosen so as to handle the effect of salinity.	Evaluation through comparison with empirical data.	Osmosis.	Handling of long-term intrusion of saline water.
BfT15. Montmorillonite transformation	Model calculations (thermal, saturated phase; unsaturated phase disregarded).	Model calculations.	Reactions in porewater and clay, sorption.	
BfT16. Colloid release	Neglected if total cation charge is > 4 mM. Otherwise modelled.	Neglected if total cation charge is > 4 mM. Otherwise modelled.	Colloid release.	Loss of backfill is discussed in Section 10.4.8.
BfT17. Radiation-induced transformations	Neglected, since dose rate in backfill is too low to have any effect.	Neglected, since dose rate in backfill is too low to have any effect.	–	
BfT18. Microbial processes	Excluded, (the effect on oxygen consumption is not considered).	Mass balance considerations.		

	Resaturation/"thermal" period	Long-term after saturation and "thermal" period	FEP chart item (see Section 8.5)	Reference to Sections in this report (if relevant) and other notes
Failed canister				
BfT5 Failed can. Gas transport/dissolution	*Neglected, since gas volumes (from buffer) assumed to be too low to reach backfill during this period.*	*Neglected, pessimistically since transport would delay radioactive releases and decrease buffer pressure. The backfill would act as a sink for gas.*	–	*Gas release from canister.*
BfT19. Colloid formation and transport	*See geosphere (no failures are expected this period).*	*See geosphere.*	–	*Called "colloid transport" for buffer. Reference to corresponding geosphere process.*
BfT20. Speciation of radionuclides	*Assumptions based on empirical data (no failures are expected this period).*	*Assumptions based on empirical data.*	–	See Chapter 13.
BfT21. Transport of radionuclides in water phase	*COMP23 (no failures are expected this period).*	*COMP23.*	–	See Chapter 13.
BfT22. Transport of radionuclides by a gas phase	*By-passed (no failures are expected this period).*	*By-passed.*	–	See Section 13.8.

7.4.5 Geosphere

Table 7-6. Process table for the geosphere describing how processes are handled in different time frames/climate domains and in the special case of earthquakes. Green fields denote processes that are neglected or irrelevant. Red fields denote processes that are quantified by modelling in the safety assessment. Orange fields denote processes that are neglected subject to a specified condition. Motives for the adopted handling are given in the Geosphere process report. An overview of all modelling activities, with references to sections in this main report where the results are discussed, is given in Section 7.5. Much of this information is also given in the last column in the table. Model acronyms are explained in Table 7-7.

Process	Excavation/operation	Temperate	Periglacial	Glacial	Earthquakes	FEP chart item (see Section 8.5)	Reference to sections in this report (if relevant) and other notes
Ge1. Heat transport	Neglected since sensitivity studies show that it takes very specific excavation/deposition sequences for heat generation to influence.	Modelling of peak canister temperature, assesment of distribution of peak temperature among the canisters and temperature distribution in rock.	Site-specific 2-D estimations of temperature distribution with depth.	Site-specific 1-D estimations of subglacial permafrost- and freezing depths.	Not relevant.	Row “Temperature”.	Temperate: Section 10.3.4. Permafrost and glacial: Section 10.4.3.
Ge2. Freezing	Not relevant.	Not relevant.	Site-specific 2-D estimations of permafrost- and freezing depths.	Site-specific 1-D estimations of subglacial permafrost- and freezing depths.	Not relevant.	Row “Temperature”.	Section 10.4.3.
Ge3. Groundwater flow	Modelling of inflow, water table drawdown, and salt water upconing assuming saturated groundwater flow using DarcyTools. MIKE SHE used for simulating water table drawdown effects in detail.	Modelling of backfill resaturation using DarcyTools and saturated groundwater flow on different scales using ConnectFlow.	Modelling of saturated groundwater flow on a super-regional scale using DarcyTools.	Modelling of saturated groundwater flow on a super-regional scale during advance and retreat of ice sheets, with and without permafrost, using DarcyTools.	Impact on groundwater flow not specifically addressed but simplified calculations of radionuclide transport carried out (see Ge24).	Row “GW flow”.	Excavation/operation: Section 10.2.3. Temperate: Section 10.3.6. Permafrost and glacial: Section 10.4.6.
Ge4. Gas flow/dissolution	Neglected based on arguments supporting the assumption of small effects of unsaturated regions on inflows to tunnels.	Included in a simplified manner in the backfill resaturation calculations. Neglected for other calculations based on the assumption that gas generated in the repository can rapidly escape through the geosphere without causing a pressure build-up.	Neglected in SR-Site based on considerations that gas that may be trapped below permafrost will have a similar effect on groundwater flow to a slightly thicker permafrost layer. Also, gas may escape through taliks if present.	Neglected in SR-Site based on the assumption that gas generated in the repository can rapidly escape through the geosphere without causing a pressure build-up (if permafrost is not present).	Not relevant.	(Row “GW flow”.)	Section 13.8.

Process	Excavation/operation	Temperate	Periglacial	Glacial	Earthquakes	FEP chart item (see Section 8.5)	Reference to sections in this report (if relevant) and other notes
Ge5. Displacements in intact rock	3DEC modelling of near-field effects of excavation of tunnels and deposition holes.	3DEC modelling of thermal stresses and deformations in the near field and in the far field.	3DEC modelling of horizontal stress reduction caused by cooling.	3DEC stress modelling of near field. ABAQUS modelling of far field.	Included in the modelling of shear movements.	Row "Rock stresses"	Excavation/operation: Section 10.2.2. Temperate: Section 10.3.5 Permafrost and glacial: Section 10.4.4
Ge6. Reactivation – displacement along existing discontinuities	3DEC modelling of construction-induced reactivation.	3DEC modelling of reactivation due to thermal load (near field). Assessment of reactivation based on 3DEC stress evolution (far field). Estimation of earthquake probability (consequence analysis, see Earthquake).	3DEC modelling of fracture reactivation caused by thermal stress reduction, forebulge stress conditions and pore overpressure under impermeable permafrost layer (near field). Assessment of reactivation based on ABAQUS stress evolution, 3DEC thermal stress reduction and pore overpressure estimates (far field). Estimation of earthquake probability (consequence analysis, see Earthquake).	3DEC modelling of ice-load induced reactivation (near field). Assessment of reactivation based on ABAQUS stress model and pore overpressure estimates. Estimation of earthquake probability (consequence analysis, see Earthquake).	Apply design rules (respect distances and canister spacing). Assessment of residual probability for canister failure due to shear displacement.	Row "Fracture structure"	See process Ge5. Integrated handling of effects of earthquakes in Section 10.4.5
	Construction-induced seismicity neglected since extraction rate is too small to generate anything but local and limited construction-induced potential instability.					Row "Fracture structure"	
Ge7. Fracturing	Assessment of EDZ. 3DEC modelling of potential for spalling. Observations (APSE) of size and shape of fractured (spalled) zone around deposition holes.	3DEC modelling of potential for spalling. Observations of size and shape of fractured (spalled) zone around deposition holes.	Thermal effects modelled and neglected provided that only marginal changes in mechanical state occur. Assessment of hydraulic fracturing under impermeable permafrost layer.	3DEC modelling of potential for fracturing induced by ice load (near field). Assessment of risk for hydraulic fracturing.	Neglected based on lack of observations at relevant distances from earthquake faults of earthquake-induced damage around open tunnels at shallow depth.	Row "Fracture structure"	See process Ge5.
Ge8. Creep	Not relevant. Covered by construction-induced reactivation.	Neglected because of insignificant convergence of deposition holes at expected rock stresses.	Neglected because of insignificant convergence of deposition holes at expected rock stresses.	Neglected because of insignificant convergence of deposition holes at expected rock stresses.	Not relevant.	–	See also Section 10.3.5.

Process	Excavation/operation	Temperate	Periglacial	Glacial	Earthquakes	FEP chart item (see Section 8.5)	Reference to sections in this report (if relevant) and other notes
Ge9. Surface weathering and erosion	The description of the handling of this process has been moved to the **Climate report**. (Index numbers of subsequent processes not changed to ensure compatibility with the **FEP report**.)						
Ge10. Erosion/sedimentation in fractures	Neglected because of too low flow rates in non-grouted fractures.	Neglected because expected hydraulic gradients and shear stresses are too low to cause significant erosion.	Neglected because expected hydraulic gradients and shear stresses are too low to cause significant erosion.	Neglected because expected hydraulic gradients and shear stresses are too low to cause significant erosion.	Not relevant.	–	–
Ge11. Advective transport/mixing of dissolved species	Advection and dispersion of salt included in saturated groundwater flow modelling using DarcyTools. Composition of mixtures assessed based on the groundwater flow modelling and site understanding.	Advection and dispersion of salt and reference waters included in saturated groundwater flow modelling using ConnectFlow. Composition of mixtures assessed based on the groundwater flow modelling and site understanding.	Advection and dispersion of salt included in saturated groundwater flow modelling using DarcyTools. Composition of mixtures assessed based on the groundwater flow modelling and site understanding.	Advection and dispersion of salt included in saturated groundwater flow modelling using DarcyTools. Composition of mixtures assessed based on the groundwater flow modelling and site understanding. Modelling of oxygen penetration based on the groundwater flow modelling, matrix diffusion and reactions with matrix minerals.	Not relevant.	Rows "GW flow and GW salinity".	Excavation/operation: Section 10.2.5. Temperate: Section 10.3.7. Permafrost and glacial: Section 10.4.7.
Ge12. Diffusive transport of dissolved species in fractures and rock matrix	Diffusion of salt between mobile and immobile groundwater included in saturated groundwater flow modelling using DarcyTools.	Diffusion of salt between mobile and immobile groundwater included in saturated groundwater flow modelling using ConnectFlow.	Diffusion of salt between mobile and immobile groundwater included in saturated groundwater flow modelling using DarcyTools.	Diffusion of salt between mobile and immobile groundwater included in saturated groundwater flow modelling using DarcyTools. Modelling of oxygen penetration based on the groundwater flow modelling, matrix diffusion and reactions with matrix minerals.	Not relevant.	(Row "GW composition").	See process Ge11.

Process	Excavation/operation	Temperate	Periglacial	Glacial	Earthquakes	FEP chart item (see Section 8.5)	Reference to sections in this report (if relevant) and other notes
Ge13. Speciation and sorption	Not relevant.	Simplified K_d-approach for modelling sorption of radionuclides. Speciation considered in the selection of K_d values.	Simplified K_d-approach for modelling sorption of radionuclides. Speciation considered in the selection of K_d values.	Simplified K_d-approach for modelling sorption of radionuclides. Speciation considered in the selection of K_d values.	Not relevant.	Row "GW composition".	See the **Data report**.
Ge14. Reactions groundwater/rock matrix	Neglected since reactions are considered to be insignificant compared with the effects of reactions with fracture-filling minerals.	Neglected since reactions are considered to be insignificant compared with the effects of reactions with fracture-filling minerals.	Neglected since reactions are considered to be insignificant compared with the effects of reactions with fracture-filling minerals.	Modelling of oxygen penetration based on the groundwater flow modelling, matrix diffusion and reactions with matrix minerals.	Not relevant.	Row "GW composition".	Glacial: Section 10.4.7, subheading "Glaciation; Redox condition"
Ge15. Dissolution/ precipitation of fracture-filling minerals	Composition of groundwater in fractures modelled based on results of the groundwater flow modelling, assumed local mineral equilibria and site understanding.	Composition of groundwater in fractures modelled based on results of the groundwater flow modelling, assumed local mineral equilibria and site understanding.	Composition of groundwater in fractures modelled based on results of the groundwater flow modelling, assumed local mineral equilibria and site understanding.	Composition of groundwater in fractures modelled based on results of the groundwater flow modelling, assumed local mineral equilibria and site understanding.	Not relevant.	Row "GW composition".	See process Ge11.
Ge16. Microbial processes	Modelling of diffusive transport of methane and hydrogen, mass balance calculations of organic matter and assessment of potential for microbial processes.	Modelling of diffusive transport of methane and hydrogen, mass balance calculations of organic matter and assessment of potential for microbial processes.	Modelling of diffusive transport of methane and hydrogen, mass balance calculations of organic matter and assessment of potential for microbial processes.	Modelling of diffusive transport of methane and hydrogen, mass balance calculations of organic matter and assessment of potential for microbial processes.	Not relevant.	Row "GW composition".	Excavation/operation: Section 10.2.5. Temperate: Section 10.3.7.
Ge17. Degradation of grout	Neglected since expected effects will occur during Temperate period.	Generic modelling of effects on chemistry of fractures and changes of hydraulic conductivity in grouting boreholes.	Not specifically handled. Extrapolation of results from Temperate period.	Not specifically handled. Extrapolation of results from Temperate period.	Not relevant.	Row "GW composition".	Temperate: Section 10.3.7

Process	Excavation/operation	Temperate	Periglacial	Glacial	Earthquakes	FEP chart item (see Section 8.5)	Reference to sections in this report (if relevant) and other notes
Ge18. Colloid processes	Neglected because of insignificant impact on geochemical conditions.	Neglected because of insignificant impact on geochemical conditions.	Neglected because of insignificant impact on geochemical conditions.	Neglected because of insignificant impact on geochemical conditions.	Impact of earth-quake on col-loidal processes not addressed, but simplified calculations of radionuclide transport carried out (see Ge24).	–	See the **Geosphere process report**.
	Impact on radionuclide transport not relevant because of intact barriers.	Bounding calculations of colloid-facilitated radionuclide transport.	Bounding calculations of colloid-facilitated radionuclide transport.	Bounding calculations of colloid-facilitated radionuclide transport.		–	–
Ge19. Formation/dis-solution/reaction of gaseous species	Composition of mixtures modelled based on the ground-water flow modelling, assumed local mineral equilibria and site understanding. This affects the concentrations of dissolved CO_2.	Modelling of diffusive transport of methane and hydrogen.	Modelling of diffusive transport of methane and hydrogen.	Included in modelling of oxygen consumption. Modelling of diffusive transport of methane and hydrogen.	Not relevant.		Glacial: Section 10.4.7, subheading “Glaciation; Redox condition”
Ge20. Methane hydrate formation	Not relevant.	Not relevant.	Neglected based on site understanding coupled with assessment of the potential for hydrate formation.	Neglected based on site understanding coupled with assessment of the potential for hydrate formation.	Not relevant.	Row “GW composition”.	–
Ge21. Salt exclusion	Not relevant.	Not relevant.	Modelling of transport of outfrozen salt.	Not relevant.	Not relevant.	Row “GW salinity”	Permafrost: Section 10.4.7, subheading “Evolution during periglacial conditions”
Ge22. Radiation effects (rock and grout)	Neglected because of too low radiation fluxes.	Neglected because of too low radiation fluxes.	Neglected because of too low radiation fluxes.	Neglected because of too low radiation fluxes.	Not relevant.	–	See **Geosphere process report**.

Process	Excavation/operation	Temperate	Periglacial	Glacial	Earthquakes	FEP chart item (see Section 8.5)	Reference to sections in this report (if relevant) and other notes
Ge23. Earth currents	Neglected since expected electrical potential fields are too small to affect groundwater flow or solute transport.	Neglected since expected electrical potential fields are too small to affect groundwater flow or solute transport.	Neglected since expected electrical potential fields are too small to affect groundwater flow or solute transport.	Neglected since expected electrical potential fields are too small to affect groundwater flow or solute transport.	Not relevant.	–	See **Geosphere process report**.
Ge24. Transport of radionuclides in the water phase	Not relevant since engineered barriers are intact.	Advection, dispersion, matrix diffusion, sorption, and radioactive decay included in integrated modelling using FARF31 and MARFA.	Advection, dispersion, matrix diffusion, sorption, and radioactive decay included in integrated modelling using FARF31 and MARFA.	Advection, dispersion, matrix diffusion, sorption, and radioactive decay included in integrated modelling using FARF31 and MARFA.	No credit taken for radionuclide retardation in the geosphere i.e. geosphere far field is short-circuited.	Not explicitly included in present version.	See Chapter 13, in particular Section 13.4.2.
Ge25. Transport of radionuclides in the gas phase	Not relevant since engineered barriers are intact.	Assessed by neglecting the geosphere as a barrier.	Assessed by neglecting the geosphere as a barrier.	Assessed by neglecting the geosphere as a barrier.	Not relevant.	Not explicitly included in present version.	Section 13.8.

7.4.6 Additional system parts

Process tables for the tunnel plugs, the central underground area, the top seal, the bottom plate in the deposition hole and the borehole seals have also been developed and are included in the **Buffer, backfill and closure process report**. Since these components are less important for safety, these process tables are not presented here in the main report. The following is a summary of how processes in the additional system parts are handled in SR-Site.

Deposition tunnel plug

The purpose of the deposition tunnel plug is to restore the hydraulic conditions in a deposition tunnel once the canisters have been installed and the tunnel has been backfilled. It is important that the plug can withstand the pressure gradient and restrict any flow of water out from deposition tunnel to the open transport tunnel. The saturation and the mechanical evolution of the plug during the operational phase are treated in the THM-modelling of the buffer and backfill. The concrete part will eventually degrade and it cannot be excluded that a more porous local volume will be formed. The mechanical effect of the intrusion of backfill into the volume is also covered in these THM-calculations. The hydraulic effect of the volume is not treated in SR-Site, since it will be sealed off with backfill on both sides. The degradation of the concrete will also give an effect on the geochemical conditions in the repository. This is treated in the modelling of the geochemical evolution.

Central area

The saturation of the central area after the closure of the repository is included in the buffer and backfill THM-modelling. After saturation most processes are of little or no concern for the repository performance. The geochemical evolution, especially the effect of reinforcements, is discussed as well as the mechanical interaction between the central area backfill and the plugs in the transport tunnels.

Top seal

The processes in the top seal are of limited concern for the performance of the repository. The saturation time is estimated in the THM-modelling. The mechanical interaction with the backfill below is not modelled specifically, but the process is handled in the same fashion as the interaction between the tunnel plug and backfill.

Bottom plate in deposition hole

The bottom plate serves no purpose apart as an aid for the installation of the canister and buffer. The bottom plate may however affect the buffer mechanically, chemically and hydraulically. The mechanical effects may occur both during the saturation phase and in the long term when the cement has degraded. This is considered in the buffer and backfill THM-modelling. The chemical effect of the degradation of the concrete is analogous to the reaction between the tunnel plug and the backfill. The degraded bottom plate will have an increased hydraulic conductivity. This is considered in the radionuclide transport calculations.

Borehole seals

The borehole seals are in many respects similar to the buffer. The processes that are of particular concern and thus treated in SR-Site are:

1. Freezing, the upper part of the seals will under all circumstances experience low temperatures.
2. Water uptake, assessment of timescale of seal hydration.
3. Homogenisation during saturation, especially the interaction between the bentonite and perforated copper.
4. Homogenisation after a local loss of bentonite – in the case of bentonite colloid formation.

The chemical interactions between bentonite and concrete are not treated specifically. The same reasoning as for the interaction between tunnel plug and backfill is utilised.

7.5 Assessment model flow charts, AMFs

To give an overview of the models used in the evaluation of repository evolution, the dependencies/interactions between them, and data used in the modelling, two assessment model flow charts, AMFs, have been developed. One AMF concerns the excavation/operation and initial temperate period, Figure 7-3 and one represents permafrost and glacial conditions, Figure 7-4.

In the AMFs, the modelling tasks are displayed as rounded rectangles, data to/results from model studies are displayed as rectangles and assessments of data for further modelling based on model output and other information are displayed as boxes. Data that are qualified in the **Data report** are indicated with blue colouration and modelling tasks presented in the **Model summary report** are indicated with yellow colouration.

The **Model summary report** describes the models represented in the AMFs. Tables 7-7 and 7-8 provide links between the processes in the process tables, the modelling activities described by the AMF and the section of the reference evolution (Chapters 10, 12 and 13) where the modelling is reported.

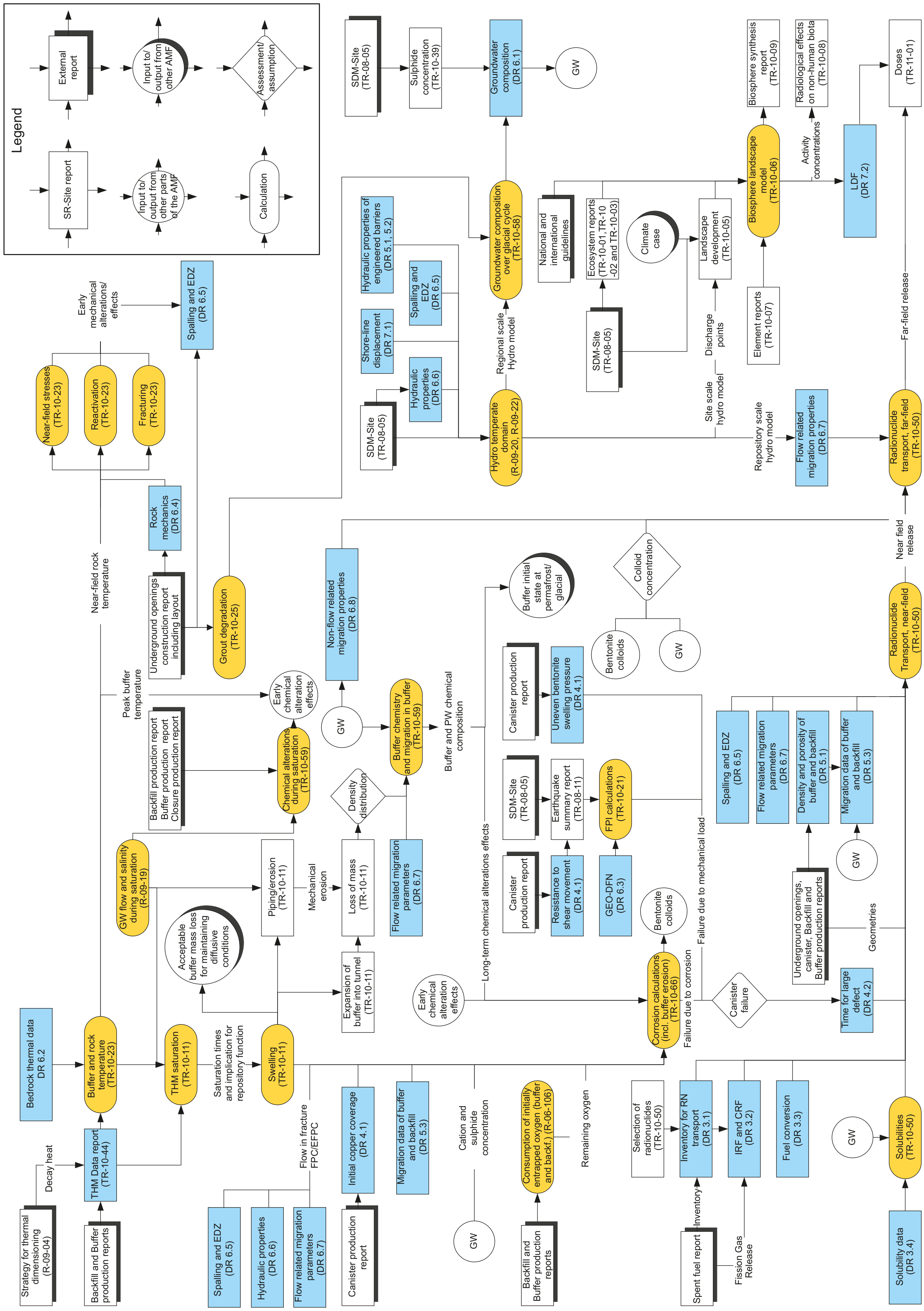


Figure 7-3. *The assessment model flow chart for the excavation/operation period and the initial temperate period after closure. See main text for further explanations.*

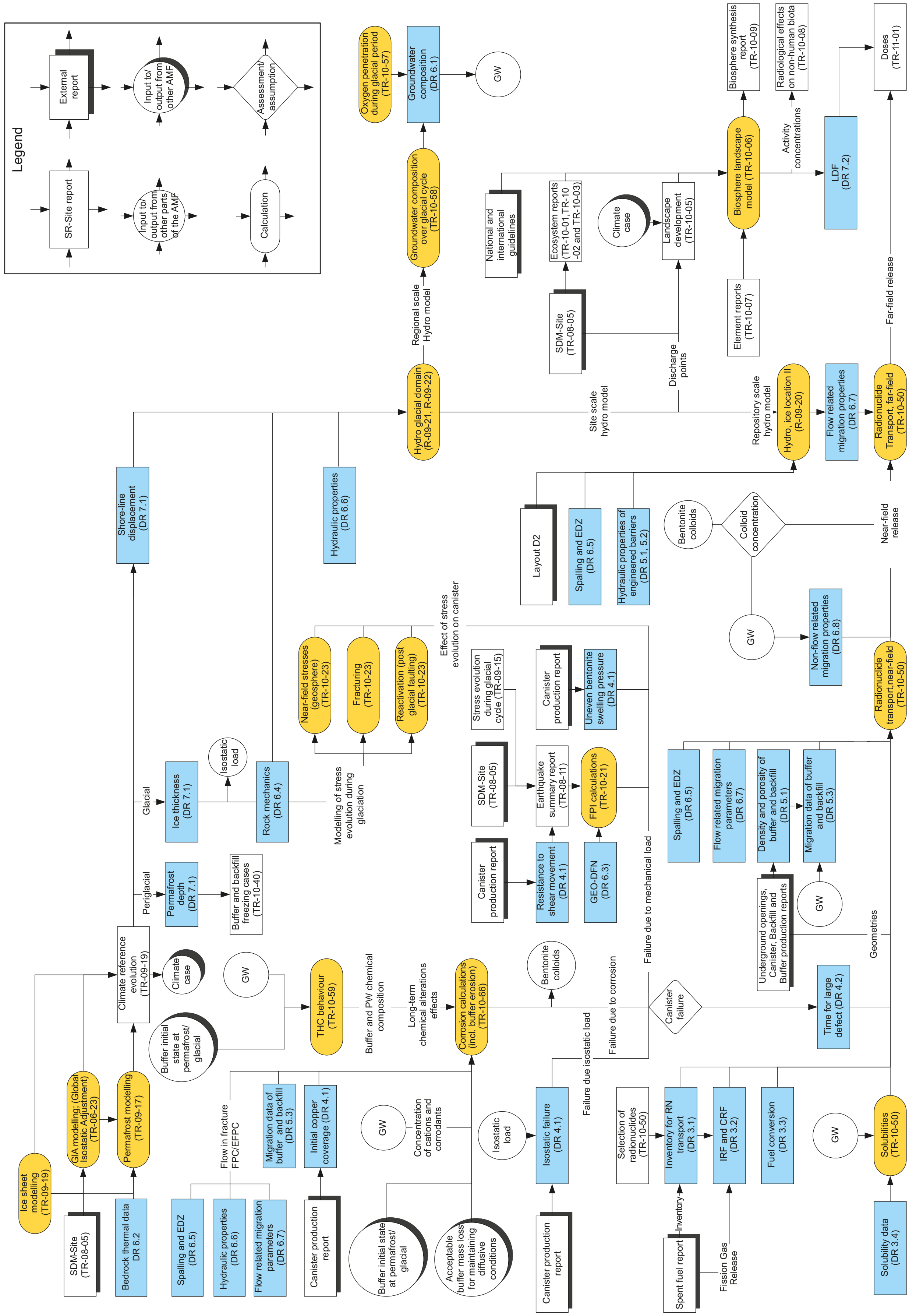


Figure 7-4. The assessment model flow chart for permafrost and glacial conditions. See main text for further explanations.

Table 7-7. Links between process tables, AMF (Figure 7-3) and reporting in this main report. Excavation/operation and temperate periods. The modelling activities in the left column correspond to yellow objects in Figure 7-3.

Modelling activity in AMF	Included processes, as indexed in process tables in Section 7.4	Code	Section(s) where modelling is reported	Note	Modelling report	
					Reference	In AMF
Buffer and rock temperature	F1, F2, Bu1, Bu2, Ge1	3DEC	10.3.4	Decay and heat generation modelled as exponential expressions fitted to results of detailed calculations	Hökmark et al. 2010	TR-10-23
THM saturation (buffer and backfill)	Bu2, Bu4, BfT3	ABAQUS	10.3.8		Åkesson et al. 2010a	TR-10-11
Near-field stresses (geosphere)	Ge5	3DEC	10.2.2 (Excavation/operation) 10.3.5 (Initial temperate)		Hökmark et al. 2010	TR-10-23
Reactivation	Ge6	3DEC	10.2.2 (Excavation/operation) 10.3.5 (Initial temperate)		Hökmark et al. 2010	TR-10-23
Fracturing (spalling)	Ge7	3DEC	10.2.2 (Excavation/operation) 10.3.5 (Initial temperate)		Hökmark et al. 2010	TR-10-23
FPI calculations; calculation of the occurrence of Full Perimeter Intersecting fractures in deposition tunnels (see Section 5.2.2)	Initial state issue	Matlab	10.4.5		Munier 2010	TR-10-21
Chemical alterations during saturation (geosphere)	Ge13, Ge15	PHAST	10.2.5		Sena et al. 2010	TR-10-59
Grout degradation	Ge17	CodeBright	10.3.7 (Initial temperate) (Mentioned also in 10.2.5 Excavation/operation)		Grandia et al. 2010a	TR-10-25
Groundwater flow and salinity during saturation	Ge3	DarcyTools	10.2.3		Svensson and Follin 2010	R-09-19
Swelling	Bu8, BfT7	ABAQUS, CodeBright	10.3.9		Åkesson et al. 2010a	TR-10-11
Buffer chemistry and migration in buffer	Bu11, Bu12, Bu13, Bu14	PHAST	10.3.10 (elevated temperatures) 10.3.12 (long-term)		Sena et al. 1020	TR-10-59
Consumption of initially entrapped oxygen (buffer and backfill)	BfT12	PHAST	10.2.5		Grandia et al. 2006	TR-06-106
Corrosion calculations (including buffer erosion calculations)	Bu18, C11	Analytical expressions (Excel)	10.2.5 (excavation/operation) 10.3.13 (initial temperate)		SKB 2010d	TR-10-66
Hydro temperate domain	BfT4, Ge3, Ge11	ConnectFlow	10.3.6		Joyce et al. 2010 Selroos and Follin, 2010	R-09-20 R-09-22
Groundwater composition over glacial cycle	Ge13, Ge14, Ge15, Ge19	PhreeqC	10.3.7 (temperate period)		Salas et al. 2010	TR-10-58

Modelling activity in AMF	**Included processes, as indexed in process tables in Section 7.4**	**Code**	**Section(s) where modelling is reported**	**Note**	**Modelling report**	
					Reference	**In AMF**
Solubilities	F14	"Simple functions"			**Radionuclide transport report**	TR-10-50
Radionuclide transport, near-field	F17, Bu25, BfT21 (The above three include, as sub-processes, F1, F12, F13, F14, Bu11, Bu12, BfT9, BfT10 and BfT11)	COMP23	Chapter 13		**Radionuclide transport report**	TR-10-50
Radionuclide transport, far-field	Ge24, consisting of sub-processes Ge11, Ge12, Ge13 and F1	FARF31 MARFA	Chapter 13		**Radionuclide transport report**	TR-10-50
Biosphere landscape model	Biosphere processes	Ecolego, MIKE_SHE, Pandora, ERICA	13.2		Avila et al. 2010	TR-10-06

Table 7-8. Links between process tables, AMF, Figure 7-4 and reporting in in this main report. Permafrost and glacial periods. The modelling activities in the left column correspond to yellow objects in Figure 7-4.

Modelling activity in AMF	Included processes, as indexed in process tables in Section 7.4	Code	Section(s) where modelling is reported	Note	Modelling report	
					Reference	In AMF
Permafrost modelling	F1, F2, Ge1	Numerical permafrost model	10.4.1, 10.4.3		In the **Climate report,** details in Hartikainen et al. 2010	In TR-10-49, details in TR-09-17
Ice sheet modelling	External processes, see SR-Site **Climate report**	UMISM	10.4.1		In the **Climate report**	In, TR-10-49, details in TR-09-19
GIA modelling; (Global Isostatic Adjustment)	External processes, see SR-Site **Climate report**	Numerical GIA model	10.4.1		In the **Climate report,** details in SKB 2006c	In, TR-10-49, details in TR-06-23
FPI calculations; calculation of the occurrence of Full Perimeter Intersecting fractures in deposition tunnels (see Section 5.2.2)	Initial state issue	Matlab	10.4.5		Munier 2010	TR-10-21
Near-field stresses (geosphere)	Ge5	3DEC	10.4.4		Hökmark et al. 2010	TR-10-23
Reactivation	Ge6	3DEC	10.4.4		Hökmark et al. 2010	TR-10-23
Fracturing	Ge7	3DEC	10.4.4		Hökmark et al. 2010	TR-10-23
Groundwater composition over glacial cycle	Ge3, Ge11, Ge12, Ge21	PhreeqC	10.4.7		Salas et al. 2010	TR-10-58
Hydro, glacial domain	Ge3, Ge11	DarcyTools	10.4.6		Vidstrand et al. 2010 Selroos and Follin 2010	R-09-21 R-09-22
Hydro, ice location II	Ge3, Ge11	ConnectFlow	10.4.6		Joyce et al. 2010	R-09-20
Oxygen penetration during glacial period	Ge11, Ge15	PhreeqC, PHAST, analytical expressions	10.4.7		Sidborn et al. 2010	TR-10-57
Buffer and backfill freezing cases	Bu3, BfT2		10.4.8		Birgersson et al. 2010	TR-10-40
THC behaviour	Bu11, Bu12, Bu13, Bu14	PHAST	10.4.8		Sena et al. 2010	TR-10-59
Corrosion calculations (including buffer erosion calculations)	Bu18, C11	Analytical expressions (Excel)	10.4.9		SKB 2010d	TR-10-66
Solubilities	F14	"Simple functions"			**Radionuclide transport report**	TR-10-50
Radionuclide transport, near-field	F17, Bu25, BfT21 (The above three include, as sub-processes, F1, F12, F13, F14, Bu11, Bu12, BfT9, BfT10 and BfT11)	COMP23	Chapter 13		**Radionuclide transport report**	TR-10-50
Radionuclide transport, far-field	Ge24, consisting of sub-processes Ge11, Ge12, Ge13 and F1.	FARF31 MARFA	Chapter 13		**Radionuclide transport report**	TR-10-50
Biosphere landscape model	Biosphere processes	Ecolego, MIKE_SHE, Pandora, ERICA	13.2		Avila et al. 2010	TR-10-06

8 Safety functions and safety function indicators

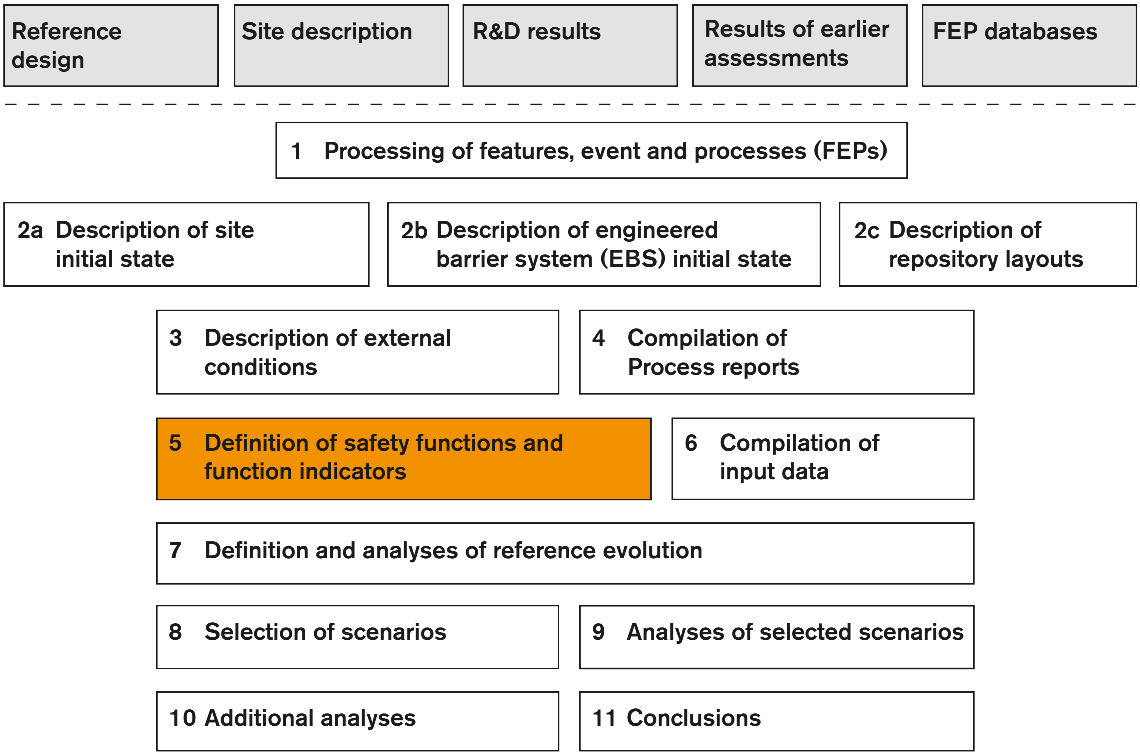


Figure 8-1. *The SR-Site methodology in eleven steps (Section 2.5), with the present step highlighted.*

8.1 Introduction

8.1.1 Differentiated safety functions in SR-Site

As mentioned in Section 2.1, the primary safety function of the KBS-3 concept is to completely contain the spent nuclear fuel within copper canisters over the entire assessment period, which is one million years in SR-Site. Should a canister be damaged, the secondary safety function is to retard any releases from the canisters. The two issues of containment and retardation are thus of primary importance throughout the assessment. It should be noted that the containment function is more prominent in the KBS-3 concept than in many other repository concepts for spent nuclear fuel or high level waste e.g. /Nagra 2002, Andra 2005/. This is also reflected in the methodology and structure of the safety assessment, which focuses to a comparatively large extent on the containing capacity of the repository.

In the safety assessment SR-Can, more differentiated safety functions for a KBS-3 repository were introduced and used to focus the assessment on important factors for long-term safety, for giving a structured account of the reference evolution and as a basis for scenario selection. The approach was seen as a positive development in the review of the SR-Can assessment /Dverstorp and Strömberg 2008/ and is also well in line with international developments in the area of safety assessment methodology /NEA 2009/.

This chapter deals with differentiation of safety functions for a KBS-3 repository for SR-Site, building on the work in the SR-Can assessment. Safety function indicators and criteria for these indicating safe conditions in the repository are introduced in Section 8.2. Safety function indicators and criteria related to containment and retardation are developed in Sections 8.3 and 8.4, respectively. In Section 8.5, a FEP chart is developed. The chart describes the connections between important initial state conditions, long-term processes and safety functions and thus integrates much of the information given in this chapter, Chapters 5 and 7, and external factors described in Chapter 6.

The safety functions and their indicators and criteria are used to structure the evaluation of safety when the long-term evolution of the repository is evaluated in Chapter 10. They also play a key role in the selection and analyses of scenarios in Chapters 11 and 12, respectively.

8.1.2 Approach to dilution

Dilution is sometimes seen as a safety function in the context of waste management. Dilution is, however, not seen as a safety function for the KBS-3 system. The main reasons for this are that dilution essentially cannot be controlled by the design of the repository and only to some extent by site selection. Nevertheless, dilution will play an essential role in a realistic estimate of the consequences of a potential release from the repository. A coastal site can be expected to be submerged for extended periods of time and dilution of potential releases in sea water could dramatically lower the calculated annual effective doses and thus the associated radiological risks.

The future evolution of climate and climate-related conditions, which will be determining factors for dilution, is however uncertain; many climate-related parameters may vary by several orders of magnitude. Although marine discharges can be predicted to exist for long periods, these will also be interrupted by periods during which releases occur to terrestrial ecosystems, or when earlier releases accumulated in sea sediments will be present in terrestrial systems due to shore-line displacement. The terrestrial conditions will likely be associated with the highest individual risks. The compliance discussion for a repository has to be based on these unfavourable but, in a long-term perspective, not unlikely conditions.

Inevitably, dilution has to be included in quantitative assessments of wells but also in this case the situations that will occur are not amenable to control by repository design or site considerations.

A related phenomenon concerns the fact that the repository is distributed in space and that the host rock and the near-surface hydrological systems will redistribute potential releases from the repository before they cause human exposure. These phenomena have to be included in the quantification of consequences of potential releases. The redistribution effects cannot however be straightforwardly described as positive or negative.

8.2 Safety functions, safety function indicators and safety function indicator criteria; general

The overall criterion for evaluating repository safety is the risk criterion issued by SSM, which states that “the annual risk of harmful effects after closure does not exceed 10^{-6} for a representative individual in the group exposed to the greatest risk”. This is a “top level” criterion that requires input from numerous analyses on lower levels, and where the final risk calculation is the integrated result of various model evaluations using a large set of input data.

Safety functions

A detailed and quantitative understanding and evaluation of repository safety requires a more elaborated description of how the main safety functions of containment and retardation are maintained by the components of the repository. Based on the understanding of the properties of the components and the long-term evolution of the system, a number of subordinate safety functions to containment and retardation can be identified.

In this context, a *safety function* is defined qualitatively as a role through which a repository component contributes to safety. For example, canisters should resist isostatic loads in the repository without the containment function being breached. A safety function related to the canister and subordinate to containment would therefore be the ability of the canister to *resist isostatic loads*.

Safety function indicators

In order to quantitatively evaluate safety, it is desirable to relate or express the safety functions to measurable or calculable quantities, often in the form of barrier conditions.

For the canister's function of resisting isostatic loads in the repository, the total isostatic load with contributions from the buffer swelling pressure and the hydrostatic pressure is a suitable quantity to use in order to evaluate the extent to which this safety function is fulfilled. The isostatic load is said to be a *safety function indicator*[14] for the mentioned canister safety function. A safety function indicator is thus a measurable or calculable quantity through which a safety function can be quantitatively evaluated.

Safety function indicator criteria

In order to determine whether a safety function is maintained or not, it is desirable to have quantitative criteria against which the safety function indicators can be evaluated over the time period covered by the safety assessment.

The situation is however different from safety evaluations of many other technical or industrial systems in an important sense: The performance of the repository system or parts thereof do not, in general, change in discrete steps, as opposed to e.g. the case of a pump or a power system that could be characterised as either functioning or not (possibly in addition to intermediate states of partial functioning). The repository system will evolve continuously and in many respects there will be no sharp distinction between acceptable performance and a failed system on a sub-system level or regarding detailed barrier features.

There are thus many safety function indicators on which no limit for acceptable performance can be given. The groundwater concentrations of canister corroding agents or agents detrimental to the buffer are examples of this kind of factor related to containment. Usually, they enter in more complex analyses where a number of parameters together determine, e.g. the corrosion rate of the canister. Most of the factors determining retardation are also of this nature.

Nevertheless, as will be demonstrated in this chapter, there are some crucial barrier properties on which quantitative limits for safe functioning can be put. Regarding containment, an obvious condition is the requirement that the copper canister should nowhere have a penetrating defect, i.e. there should, over the entire surface of the canister, be a non-zero copper thickness. In addition to this direct measure of containment performance, a number of quantitative supplementary criteria can also be defined. These relate, for example, to the peak temperature in the buffer and to requirements on buffer density and buffer swelling pressure giving favourable buffer properties for maintaining containment. Most of them determine whether certain potentially detrimental processes can be excluded from the assessment. Relating to the above example of isostatic loads in the repository, the design analysis of the canister has demonstrated that the canister withstands an isostatic load of 45 MPa. The requirement that the isostatic load should not exceed 45 MPa is thus a *safety function indicator criterion* in this case.

Relation between global safety and individual safety functions

It is emphasised that the breaching of a safety function indicator criterion does not mean that the repository is unsafe, but rather that more elaborate analyses and more extensive data are needed in order to evaluate safety.

The criteria are an aid in determining whether safety is maintained. If the criteria are fulfilled, the safety evaluation is facilitated. If all criteria related to canister failures are fulfilled, this implies that the overall risk criterion is fulfilled, provided that all canister failure modes have been identified. Fulfilment of all criteria related to the buffer, the backfill and the host rock is not a guarantee for compliance with the overall risk criterion, since the canister could still be failed so that releases of radionuclides could occur. On the other hand, compliance with the risk criterion could well be compatible with a violation of one or several of the criteria. A violation would be an implication of caution; further analyses could be required in order to determine the consequences on a sub-system level or a system level.

[14] In choosing the term "function indicator", it was observed that the two terms "performance indicator" and "safety indicator" in this context normally refer to releases of radionuclide or resulting dose consequences /Becker et al. 2002/. Those terms were thus avoided.

An example is the criterion that the charge concentration of groundwater cations should exceed 4 mM, $\Sigma q[M^{q+}] > 4$ mM, in order for buffer erosion to be excluded. If this criterion is breached, buffer erosion must be quantitatively evaluated and its consequence, in terms of reduced buffer density, needs to be propagated to assessments of buffer swelling pressure and hydraulic conductivity. Alterations of the latter factors could, in turn, influence e.g. canister corrosion. A chain of assessments is thus initiated by the breaching of the first safety function, but the final outcome of a possibly increased corrosion rate is not necessarily an unacceptable impact on containment.

Approach to margins

Related to the above, a criterion may be defined so that it includes a considerable margin to unacceptable performance.

The peak temperature criterion for the buffer, set to 100°C in order to avoid mineral transformations, is an example of a criterion with a considerable margin as documented in the **Buffer, backfill and closure process report**, Section 3.5.9. One reason for this is that the extent of mineral transformation increases gradually with temperature and it is not possible to determine a sharp limit below which no transformation occurs. Rather, a criterion is determined below which transformation can, for all practical purposes and for long time spans, be neglected in the safety assessment.

An example of a criterion with a smaller margin is the requirement that the total groundwater charge concentration of cations should exceed 4 mM in order to rule out erosion/colloid formation of the buffer. Here, a sharper onset of the safety function exists and it is possible to formulate the criterion with a smaller margin.

For the safety function indicator criteria used in SR-Site, there has been no systematic approach to margins when determining the criteria. The only requirement applicable to all safety function indicator criteria is that the safety function to which it relates should be fulfilled if the criterion is satisfied, based on the scientific understanding of the phenomenon in question. For application in the safety assessment, the margin for each indicator is of interest. Therefore, discussions of margins for all defined indicator criteria related to containment and retardation are given in Sections 8.3.5 and 8.4.6, respectively.

Quantities for safety function indicators

There is, for some safety functions, a certain degree of freedom in the choice of quantities for the indicators used to represent the safety function.

For example, in the presently developed version, the indicator used to quantify the buffer safety function “prevent colloid transport through buffer” is the buffer density, whereas one could also have chosen the buffer pore size, a more direct measure of the safety function. For a specific bentonite material, the pore size is however directly related to the density and the buffer density is of interest in many other aspects of the safety assessment. Therefore, the density was chosen as the safety function indicator in this case.

There are other similar examples, in particular for the buffer for which many characteristics are dependent and thus to some degree interchangeable.

Derivation of safety functions, indicators and criteria

For the set of safety functions, their indicators and criteria to be useful in the evaluation of safety, they need to be sufficiently comprehensive. It is therefore important to have a systematic approach to the derivation of these entities.

The pillars on which the derivation of safety functions is built are:

- The two principal safety functions containment and retardation on which the design of the KBS-3 repository is based.
- The scientific understanding of the long-term evolution of a KBS-3 repository.

Throughout the decades of research related to the long-term safety of a KBS-3 repository, safety functions or barrier requirements have been discussed and established successively.

In SR-Site, the results of these efforts have been utilised. Also, all canister and buffer processes identified as relevant for long-term safety and documented in the **Process reports** have been considered with the aim of determining if a safety function relating to the process could be defined, ideally accompanied by an indicator and a criterion, see further Sections 8.3.1 and 8.3.2, sub-headings "Additional considerations".

As for the set of processes identified as important for long-term safety, completeness can never be unequivocally claimed for the set of safety functions in the evaluation of safety. The set of safety functions can be more or less mature, as a reflection of the maturity of the scientific understanding of the system analysed. The safety of the KBS-3 system has been studied for decades and new detrimental processes, that could form a basis for the formulation of additional safety functions, have not been identified in recent years. Furthermore, the principle of designing a relatively simple system using naturally occurring materials with well known long-term properties, as in the case of KBS-3, favours the derivation of a comprehensive and mature set of safety functions.

Safety function indicator criteria are not the same as design premises

It is noted that safety function indicator criteria are not the same as design criteria, formalised into design premises as discussed in Chapter 5. Safety function indicator criteria are meant to be fulfilled throughout the one million year assessment period, whereas design premises relate to the initial state of the repository. Design premises need to be defined with sufficient margin to allow deterioration of the system components over the assessment period so that safety is still fulfilled, i.e. so that, ideally, all the safety function indicator criteria are fulfilled also at the end of the assessment period. A clear example of this is the copper thickness of the canister: It is designed with a 5 cm copper thickness to allow for corrosion, whereas the safety function indicator criterion requires that it is nowhere zero, as this is the criterion for the breaching of containment.

There are also a number of design premises relating to limits on content of detrimental agents in the engineered components of the repository, e.g. the phosphorous and oxygen contents of the copper and the pyrite content of the buffer. These properties are determined by the production and control procedures relating to the component in question and given as initial state conditions for the safety assessment. They need to be included in the analysis of relevant detrimental processes in the assessment; for the three above examples this would be in the analysis of canister corrosion. They are, however, not necessarily suited as safety function indicators since they do not change over time other than by being consumed in the analysed detrimental processes.

The situation is, however, different for the corresponding safety function indicators and criteria related to future geosphere conditions, e.g. groundwater concentrations of solutes detrimental to the buffer or the canister. These can only to a limited extent be controlled by design or siting choices and are thus more relevant to include as function indicators for long-term safety.

Relations between safety functions

The safety functions are related. All safety functions of the buffer either support a safety function of the canister, or contribute to retardation in the buffer. For example, the safety function "limit advective transport" in the buffer supports the canister safety function "provide corrosion barrier", and also contributes to retardation in the buffer since advection is a more efficient transport mechanism than diffusion. Similarly, all safety functions of the host rock either support a safety function of the canister directly or indirectly via a buffer safety function, or contribute to retardation in the rock.

Summary

The following definitions have been introduced:

- A safety function is a role through which a repository component contributes to safety.
- A safety function indicator is a measurable or calculable property of a repository component that indicates the extent to which a safety function is fulfilled.
- A safety function indicator criterion is a quantitative limit such that if the safety function indicator to which it relates fulfils the criterion, the corresponding safety function is maintained.

Safety functions are an aid in the evaluation of safety, but the fulfilment of all safety function indicator criteria is neither necessary nor sufficient to argue safety. The different safety function indicator criteria are furthermore determined with varying margins to acceptable performance.

Safety function indicator criteria are related to, but not the same as, design premises. Whereas the latter relate to the initial state of the repository and primarily to its engineered components, the former should be fulfilled throughout the assessment period and relate, in addition to the engineered components, to the natural system.

The set of safety functions used in SR-Site has been derived based on the documented experience accumulated over decades of research related to the long-term safety of the KBS-3 repository.

8.3 Safety functions for containment

Safety functions, function indicators and, where applicable, function indicator criteria for containment are presented below. All defined entities are summarised in Figure 8-2 at the end of this section using the function labelling introduced in the subheadings in the following sections (Can1, Can2 etc.). The criteria presented below are often selected from a cautious perspective and further studies and engineering development may show that some of the criteria could be relaxed for future safety assessments. Also, additional criteria may be added.

8.3.1 Canister

Can1. Provide corrosion barrier

The canister integrity can be threatened either mechanically or chemically. An obvious requirement regarding canister integrity is that the copper shell of the canisters should not be penetrated. This can be expressed such that the minimum copper thickness taken over the entire canister surface shall be larger than zero:

$$d_{\min}^{Cu} > 0$$

As long as this criterion is strictly fulfilled for all canisters, containment is complete and no releases occur.

Can2. Withstand isostatic load

A key safety function of the canister is its ability to withstand isostatic loads in the repository.

Maximum isostatic pressures in a repository were, in the SR-Can assessment, estimated to not exceed 45 MPa, composed of the bentonite swelling pressure and the hydrostatic pressure at the disposal depth during a glaciation /SKB 2006a/. Therefore, it is a design premise that the canister should withstand a 45 MPa isostatic load, see further Section 5.4.1. The evaluation of the design and manufacturing procedures for the ensemble of deposited reference canisters led to the conclusion that they will fulfil this design premise, Section 5.4.3.

The isostatic load on the canister is, therefore, used as a safety function indicator in SR-Site and the design premise of 45 MPa is used as the corresponding safety function indicator criterion. This does, however, not necessarily mean that a canister will fail if the load exceeds 45 MPa, since, in the design analyses of the canister, pessimistically chosen criteria are used when assessing the results of the underlying strength calculations and damage tolerance analyses. This is reflected by the results of two pressure tests of canister mock-ups where collapse occurred around 130 MPa /Raiko et al. 2010/.

In SR-Site, it is, therefore, strictly assumed that canisters withstand isostatic loads up to 45 MPa, based on the results of the design analysis. For potential loads exceeding 45 MPa, a more detailed scrutiny of the results of the design analysis is required.

Can3. Withstand shear load

Another key safety function of the canister is its ability to withstand shear movements in fractures intersecting the canister's deposition hole.

The design premise related to shear movements is given in Section 5.4.1: "The copper corrosion barrier should remain intact after a 5 cm shear movement at a velocity of 1 m/s for buffer material properties of a 2,050 kg/m^3 Ca-bentonite, for all locations and angles of the shearing fracture in the deposition hole, and for temperatures down to 0°C. The insert should maintain its pressure-bearing properties to isostatic loads."

The evaluation of the design and manufacturing procedures for the ensemble of deposited reference canisters led to the conclusion that they will fulfil this design premise, Section 5.4.3.

No safety function indicator criterion for the canister relating to shear movements is formulated since no further detailed mechanical analyses of the canister are carried out in the safety assessment. Rather, the findings in the design analysis of the canister are adopted and lead to the following safety function indicators and criteria on other parts of the system:

- For the buffer, that its density should not exceed 2,050 kg/m^3.
- For the geosphere, that the magnitude of shear movements in fractures intersecting deposition holes should not exceed 5 cm and that their velocity should not exceed 1 m/s. Furthermore, the temperature in the repository should not be below 0°C when such shear movements occur for the calculations in the design analysis to be strictly applicable.

If these conditions are fulfilled, then it is assumed in the safety assessment that the canister will not fail due to shear movements. Should any of the conditions be violated, a more detailed scrutiny of the results of the design analysis is required.

Additional considerations

A brief examination of all long-term canister processes was carried out to check whether any additional safety functions could be identified based on processes of relevance for long-term safety. The following resulted.

- The canister attenuates the radiation from the fuel and thereby reduces the radiation levels exterior to the canister (process C1 in Table 7-3). A function relating to this property could in principle be defined. This was not done since the attenuation function is fulfilled initially and is therefore trivially fulfilled in the long-term as long as the canister is intact (radiation levels decrease rapidly with time whereas the extent of attenuation in the canister is intact). The radiation effects outside the canister are related to processes that impact on the containment function, meaning that they are essentially irrelevant for a failed canister.
- The canister transports heat from its interior to the exterior (process C2 in Table 7-3). Efficient heat transfer ensures limited peak temperatures in the canister interior, whereas the heat transfer does not influence the exterior temperatures, as these are determined by properties of the buffer and the rock and also by the heat output from the waste. A function relating to this property could in principle be defined. This was not done since peak temperatures in the canister interior are reached typically ten years after deposition, a time during which the heat transfer properties of the canister do not change. It is therefore not relevant to use this property as a long-term safety function.
- The following processes are related to safety functions:
  - C3 "Deformation of cast iron insert" through functions Can2 and Can3 (see Figure 8-2).
  - C4 "Deformation of copper canister from external pressure" through functions Can2 and Can3.
  - C6 "Copper deformation from internal corrosion products" through function Can4, defined in Section 8.4.2 and Figure 8-3.
  - C8 "Corrosion of cast iron insert" through function Can4.
  - C9 "Galvanic corrosion" through function Can4.
  - C11 "Corrosion of copper canister" through function Can1.
  - C15 "Radionuclide transport" through function Can4.

- The following processes have not led to the definition of safety functions since they were deemed as insignificant for long-term safety in the evaluation in the **Fuel and canister process report**:
  - C5 "Thermal expansion (both cast iron insert and copper canister)".
  - C7 "Radiation effects".
  - C10 "Stress corrosion cracking of cast iron insert".
  - C12 "Stress corrosion cracking, copper canister".
  - C13 "Earth currents – stray current corrosion".
  - C14 "Deposition of salts on canister surface".

  For these processes, the following is noted:

  - The basis for neglecting these processes is revisited in Section 14.4 where it is sought to verify that FEPs omitted in earlier parts of the assessment are negligible in light of the completed scenario and risk analysis.
  - None of these processes, even if they were to have a significant impact on safety, would generate new safety functions. They would rather lead to the introduction of additional safety function indicators and possibly criteria for these for already existing functions. For example, the process C14 "Deposition of salts on canister surface" has a (negligible) impact on the existing function Can1 "Provide corrosion barrier". Had it been deemed important for long-term safety, then it would be evaluated in the following analyses together with other processes threatening the function Can1.

As long as the containment is intact, the possibility of criticality is ruled out. Therefore, no safety function related to criticality is formulated for an intact canister. See further Section 8.4.

There are also a number of design premises related to the material composition of the copper shell and of the cast iron insert. These include limits on the oxygen and phosphorous contents in the copper and on the copper content in the cast iron. All these design premises are evaluated in the determination of the initial state of the repository and the results of that evaluation are then used as premises in the analysis of the long-term evolution. Their values do not change in time and they are thus not suitable as indicators of safety. In addition, all these premises are related to processes that impact on existing functions.

8.3.2 Buffer

Buff1. Limit advective transport

An important safety function of the buffer is to limit transport of dissolved copper corroding agents to the canister and potential radionuclide releases from the canister. The material of the buffer surrounding the canister has been chosen so as to prevent advective transport in the deposition hole. A guideline is that the hydraulic conductivity of the buffer should fulfil, see the **Buffer, backfill and closure process report**, Section 3.3.2:

$k^{Buff} < 10^{-12}$ m/s

The requirement refers to all parts of the buffer, i.e. the variability within the buffer must be such that the requirement is everywhere fulfilled.

For any reasonable hydraulic gradient in the repository, this condition will mean that transport in the buffer will be dominated by diffusion. The hydraulic conductivity is strongly related to the density of the buffer, to the adsorbed ionic species and to the ionic strength of the surrounding groundwater.

The buffer homogeneity is ensured partially by the fact that the buffer is made of a clay material that swells when water saturated. A swelling pressure criterion is therefore formulated, see the **Buffer, backfill and closure process report**, Section 3.4.1:

$P_{Swell}^{Buff} > 1$ MPa

The requirement refers to all parts of the buffer, i.e. the variability within the buffer must be such that the requirement is everywhere fulfilled.

Diffusion controlled transport in the buffer in combination with the buffer being in tight contact with the wall of the deposition hole, which is obtained if the swelling pressure criterion is fulfilled, contributes to increasing the transport resistance in the buffer/rock interface, see further Section 8.3.4.

Buff2. Reduce microbial activity

The sulphide production by sulphate reducing bacteria present initially in the buffer is, in the long-term, normally bounded to insignificant levels by their reliance on nutrients present in the groundwater.

In certain transient situations, the access to nutrients could be significant, e.g. due to degradation of construction and stray materials in the repository. In such cases, the buffer has the function of reducing the activity of initially present microbes. Result from studies with compacted bentonite performed at *in situ* conditions at the Äspö Hard Rock Laboratory indicate that with a buffer density of 1,800 kg/m^3 the sulphide produced would lead to less than 2 micrometers of copper corrosion in one thousand years if the microbes are given free access to nutrients, see the **Fuel and canister process report**, Section 3.5.4. The activity decreases with increasing density. The quantitative treatment of a situation of this type would, however, depend on a number of factors, meaning that while the buffer density is a useful indicator for this buffer function, a strict criterion on buffer density cannot be formulated. (Density variations in the buffer should also be taken into account.)

The above concerns microbes initially present in the buffer. To prevent additional microbes from intruding, a buffer density much less than the reference density (1,950–2,050 kg/m^3) is sufficient.

Buff3. Damp rock shear movements

Another safety function of the buffer is to protect the canister from rock movements, in particular from the consequences of rock shear movements. Also here the buffer density plays a critical role, and, as mentioned in Section 8.3.1, the following design premise has been established:

$\rho_{Bulk}^{Buff} < 2{,}050$ kg/m^3 (Ensure protection of canister against rock shear)

Buff4. Resist transformations (requirement on temperature)

The buffer temperature should not exceed 100°C in order to limit chemical alterations, see the **Buffer, backfill and closure process report**, Section 3.5.9.

$T^{Buffer} < 100°C$

Buff5. Prevent canister sinking

Also, the swelling pressure should be sufficient to prevent the canister from sinking in the deposition hole since this would render the canister in direct contact with the rock (or the concrete bottom plate in the deposition hole) thus short-circuiting the buffer.

The main determinant of the creep rate and the resulting canister sinking is the magnitude of the mobilised shear strength (shear stress divided by shear strength), which results in an increased canister sinking. The shear strength decreases with decreasing swelling pressure. Analyses /Åkesson et al. 2010a/ of canister sinking in a deposition hole for a range of buffer densities and hence swelling pressures indicate that the total sinking will be less than 2 cm for swelling pressures down to 0.1 MPa, see further the **Buffer backfill and closure process report**, Section 3.4.1. Based on these calculations, the following safety function indicator criterion is cautiously formulated:

$P_{Swell}^{Buff} > 0.2$ MPa (Prevent canister sinking).

Buff6. Limit pressure on canister and rock

a. Swelling pressure limit

The design premise isostatic load on the canister has been determined under the assumption that the buffer swelling pressure will not exceed 15 MPa. This is the swelling pressure of a saturated buffer of density 2,050 kg/m^3 for a pessimistically chosen ion-free groundwater composition. This swelling pressure limit is thus set as a function indicator criterion for the buffer

$P_{Swell} < 15$ MPa

b. Buffer freezing

If the buffer freezes, development of damaging pressures due to expanding water cannot be ruled out. Therefore, the buffer temperature should not fall below the freezing temperature of a water-saturated buffer. The minimum buffer temperature will occur at the buffer/rock interface, therefore the limit is applied to this boundary. As the heat released from the canisters may, depending on the elapsed time after closure, play a role, it is necessary to consider deposition holes at the edge of the deposition area where the temperature will be the lowest. It has been experimentally verified that compacted bentonite does not change its sealing properties under unfrozen conditions after repeated cycles of freezing and thawing /Birgersson et al. 2010/.

If the groundwater in the rock around the buffer freezes, further cooling of the buffer decreases the swelling pressure by approximately 1.2 MPa/°C. At a critical temperature T_c, the swelling pressure is completely lost. T_c depends on the swelling pressure at 0°C. When the buffer temperature is below the critical temperature T_c ice starts forming in the buffer. T_c is thus the temperature at which freezing is initiated, whereas complete freezing occurs at much lower temperatures. For a typical buffer with a density in the interval of 1,950–2,050 kg/m^3, T_c is is in the interval −4 to −11°C, see further the **Buffer, backfill and closure process report**, Section 3.2.2. The temperature −4°C is, therefore, used as a safety function indicator criterion:

$T^{Buffer} > -4$°C

In summary, the pressure decreases from the freezing point of water surrounding the buffer down to the critical temperature. Below the critical temperature the water within the buffer may start to freeze. If the buffer freezes, the pressure exerted on the canister and the rock may increase, an issue requiring separate analyses.

Other requirements

The content of canister corroding agents in the buffer should be low. Apart from unavoidable initial amounts of oxygen, the pyrite content could pose a long-term problem, as pyrite, if not oxidised by initially present or intruding oxygen, will release sulphide, a canister corroding agent. There is, however, no absolute criterion placed on this amount; the corrosion effects of measured amounts will have to be evaluated quantitatively. As pyrite could also act as a scavenger for any initially present or intruding oxygen in the repository, the evaluation of the effects of the presence of this material in the buffer is complex.

It is also noted that the design premises for the buffer include a limit on the sulphide content of 0.5 weight percent of the total mass, corresponding to approximately 1% of pyrite.

Additional considerations

A brief examination of all long-term buffer processes was carried out to check whether any additional safety functions could be identified based on processes of relevance for long-term safety, in the same manner as described for the canister in Section 8.3.1. The following resulted:

- The water saturation of the buffer described in the process "Water uptake and transport for unsaturated conditions", process Bu4 in Table 7-4 is a pre-requisite for the safety function "Limit advective transport" to be fulfilled. However, as long as the buffer is unsaturated the safety function is not needed. Hence, no separate safety function can be defined for this process.

- The process Bu11 "Diffusive transport of species" controls both the corrosion rate of the canister and the possible transport of radionuclides. However, the key issue is if diffusion is the dominating transport mechanism, which is covered by the safety function "Limit advective transport". No separate safety function for the diffusion process itself has therefore been defined.
- The process "Sorption (including ion-exchange)" is important for the retardation of radionuclides from a failed canister. This process has the potential to generate a safety function. This is further discussed in Section 8.4.3.
- The processes "Alterations of impurities" and "Aqueous speciation and reactions" are critical for the determination of the chemistry in the buffer. In this way they will affect a number of other processes. However, there is no direct impact on the safety from these processes and no safety function has been defined.
- The following processes are related to safety functions:
  - "Heat transport" through function Buff4 (see Figure 8-2).
  - "Freezing" through functions Buff6 and R4.
  - "Water transport under saturated conditions" through function Buff1.
  - "Piping" through function Buff1(indirectly).
  - "Gas transport/dissolution" through the retardation function Buff9 (Figure 8-3).
  - "Swelling/mass redistribution" through functions Buff1, Buff2 and Buff5.
  - "Advective transport of species" through function Buff1.
  - "Montmorillonite alteration" through functions Buff4 and R1.
  - "Colloid formation though" function R1.
  - "Microbial processes" through function Buff2.
  - "Colloid filtration" through the retardation function Buff7 (Figure 8-3).
  - "Cementation" includes "stiffening" through Buff3.
- The following processes have not led to the definition of safety functions since they were deemed as insignificant for long-term safety in the evaluation in the **Buffer, backfill and closure process report**:
  - "Radiation attenuation/heat generation".
  - "Radiolysis of porewater".
  - "Radiation-induced transformations".
  - "Liquefaction"

  For these processes, the following is noted:

  - The basis for neglecting these processes is revisited in Section 14.4 where it is sought to verify that FEPs omitted in earlier parts of the assessment are negligible in light of the completed scenario and risk analysis.
  - None of these processes, even if they were to have a significant impact on safety, would generate new safety functions. They would rather lead to the introduction of additional safety function indicators and possibly criteria for these for already existing functions.

There are also a number of design premises related to the material composition of the buffer. These include a limit on the sulphide, total sulphur and organic content. All these design premises are evaluated in the determination of the initial state of the repository and the results of that evaluation are then used as premises in the analysis of the long-term evolution. Their values do not change in time and they are thus not suitable as indicators of safety. In addition, all these premises are related to processes that impact on existing functions.

8.3.3 Backfill in deposition tunnels

BF1. Counteract buffer expansion

The buffer swelling will cause an upward expansion with a resulting compression of the backfill. This needs to be counteracted by the backfill in order to keep the buffer density within the desired limits. The mechanical interaction between the buffer and the backfill is dependent on many factors. The process is treated in the **Backfill production report** and in the THM modelling of the buffer, backfill and other components /Åkesson et al. 2010a/. This function is evaluated as part of the analysis of the buffer's long-term ability of limiting advective transport, which includes studies of the buffer's density loss through its upward swelling into the backfill. The buffer indicators relating to limiting advective transport are thus used to evaluate this backfill function. In general terms, it is advantageous if the backfill density is high for this function to be fulfilled, but since there are many factors affecting this process, no single criterion can be defined.

The concentration of canister-corroding agents in the backfill should be low. As for the buffer, a certain amount of initially entrapped oxygen is unavoidable in the backfill, and the pyrite concentration could pose a long-term problem. There is, however, no specific constraint placed on this concentration; the corrosive effects of the measured concentrations will have to be evaluated quantitatively.

8.3.4 Geosphere

Many aspects of the host rock safety functions cannot generally be captured by simple criteria but require more complex analyses where the combined effect of a number of factors determine the outcome. Still it is possible to identify conditions that should be favourable with regard to containment, as well as retardation, and conditions that would ultimately be detrimental to the safety functions of the engineered barriers. This was, for example, discussed in the report on geoscientific suitability indicators and criteria for siting and site evaluation /Andersson et al. 2000/ and additional conclusions were made in the SR-Can main report /SKB 2006a/ Sections 13.6 and 13.7.

In the following, host rock safety functions and function indicators relating to chemical, mechanical, hydrogeological and thermal conditions are discussed qualitatively and quantitative limits on indicators are provided where possible.

R1. Provide chemically favourable conditions

Several characteristics of the groundwater composition are essential for providing chemically favourable conditions for the repository.

Reducing conditions

A fundamental requirement is that of reducing conditions. A necessary condition is the absence of dissolved oxygen, because any evidence of its presence would indicate oxidising conditions. The presence of reducing agents that react quickly with O_2, such as Fe(II) and sulphide is sufficient to indicate reducing conditions. Other indicators of redox conditions, like negative redox potential, are not always well defined and thus less useful as a basis. Nevertheless, redox potential is a measure of the availability of all kinetically active oxidising species.

This requirement ensures that canister corrosion due to oxygen dissolved in the groundwater is avoided. Furthermore, should a canister be penetrated, reducing conditions are essential to ensure a low dissolution rate of the fuel matrix, to ensure favourable solubilities of several radioelements and, for some elements, also to ensure redox states favourable for sorption in the buffer, the backfill and the host rock.

In addition to dissolved O_2, other oxidising groundwater components could be considered, for example nitrate and sulphate. However nitrate and sulphate can only be reactive by the intervention of microbes, which require both nutrients and reduced species such as dissolved hydrogen, methane or organic matter in order to be able to reduce nitrate or sulphate, whereas dissolved oxygen may react directly e.g. with the copper canister or the spent fuel.

Ionic strength, salinity

The salinity of the groundwater should neither be too high, nor too low. The total charge concentration of cations should exceed 4 mM in order to avoid colloid release from buffer and backfill, hence, (see the **Buffer, backfill and closure process report**, Section 3.5.11):

$\Sigma q[M^{q+}]^{GW} > 4$ mM

A criterion based on charge equivalents and not on separate concentrations for different ionic species has the advantage that the effect of ion exchange equilibrium is incorporated in a single criterion. Furthermore, modelling by /Neretnieks et al. 2009/ shows that during the transient of ion exchange, the concentration of Ca^{2+} in the seeping water drops at the bentonite-groundwater interface, whereas charge neutrality requires that the equivalent charge concentration remains constant.

Groundwaters of high ionic strengths would have a negative impact on the buffer and backfill properties, in particular on the backfill swelling pressure and hydraulic conductivity. In general, ionic strengths corresponding to NaCl concentrations of approximately 35 g/L (0.6 M NaCl) are an upper limit for maintaining backfill properties whereas the corresponding limit for the buffer is around 100 g/L (1.7 M NaCl). The limit of tolerable ionic strength is, however, highly dependent on the material properties of these components (see Section 5.5.3 and, for details, /Karnland et al. 2006/).

Colloid concentrations

The concentration of natural colloids should be low to avoid transport of radionuclides mediated by colloids. The stability of colloids is much decreased if the charge concentration of cations exceeds some millimol per litre. The condition discussed above for the stability of the buffer and backfill ($\Sigma q[M^{q+}]^{GW} > 4$ mM) is therefore also sufficient to keep the concentration of colloids suspended in groundwaters to a low level.

Concentrations of detrimental agents

Regarding canister corrosion, there should be low groundwater concentrations of other canister-corroding agents, in particular sulphide, HS^-. In addition, the groundwater should also have low concentrations of nutrients that may be used by sulphate reducing bacteria to produce sulphide. These are dissolved hydrogen, methane and organic carbon. For sulphide in the groundwater to pose a problem, earlier assessments demonstrated that, for an intact buffer, considerably higher concentrations than have ever been observed in Swedish groundwaters would be required. The quantitative extent of such corrosion also depends on the groundwater flow around the deposition hole and on the transport properties of fractures intersecting the hole.

Furthermore, low groundwater concentrations of agents detrimental to long-term stability of the buffer and backfill, in particular potassium and iron, are desirable, see the **Buffer, backfill and closure process report**, Section 3.5.10.

pH

Regarding pH, a criterion can be formulated from the point of view of buffer and backfill stability, see the **Buffer, backfill and closure process report**, Section 3.5.9:

$pH^{GW} < 11$

This is fulfilled for any natural groundwater in Sweden. However, construction and stray materials in the repository, in particular concrete, could contaminate the groundwater such that high pH values are reached.

Avoiding chloride assisted corrosion

A further requirement is that the combination of low pH values and high chloride concentrations should be avoided in order to exclude chloride assisted corrosion of the canister. In quantitative terms, the requirement is assigned the following criterion:

$pH^{GW} > 4$ and $[Cl^-]^{GW} < 2$ M

The basis for this criterion is documented in the **Fuel and canister process report**, Section 3.5.4.

R2. Provide favourable hydrogeologic and transport conditions

For the host rock to provide favourable hydrogeologic and transport conditions, the flow-related transport resistance (F) of flow paths leading into and out of the repository should be sufficiently high. Limited transmissivity of the water conducting fractures in combination with low hydraulic gradients yields limited flows and high transport resistance in the water conducting fractures.

Furthermore, the geosphere has an important function in controlling the transport resistance in the buffer/rock interface. This property is dependent on three factors: i) diffusive conditions in the buffer, ii) limited flow in the rock fractures intersecting the deposition hole, and iii) a favourable (limited) intersection area over which the exchange of solutes can occur. The first two factors are expressed by the safety functions relating to transport conditions in the buffer and the rock expressed above. The third factor is obtained by i) an intact buffer in tight contact with the wall of the deposition hole, which, in turn, is achieved through the buffer swelling pressure, and ii) limited aperture in the fractures intersecting the deposition hole. The latter factor can increase considerably through thermally induced spalling of the rock wall of the deposition hole. A suitable indicator for this safety function is the equivalent flow rate, Q_{eq}, which is an integrated measure of all the above factors.

In summary, for the host rock to have favourable hydrogeologic and transport conditions, it should have:

a) High flow-related transport resistance, F, in flow paths (connected fractures) from the surface environment to the repository and in flow paths leading away from the repository.
b) Low equivalent flow rate in the buffer/rock interface, Q_{eq}.

On neither of these indicators is it possible to put quantitative limits, but they require integrated analyses of site specific hydrogeologic conditions and evolution. (As rules-of-thumb, F-values above 10^4 yr/m and Q_{eq}-values below 10^{-4} m^3/yr can be regarded as favourable.)

R3. Provide mechanically stable environment

The mechanical stability of the host rock cannot, in most respects, be simply evaluated. However, two main reasons for potential mechanical failure of the canisters can be identified. These are collapse due to isostatic load and failure due to earthquakes causing secondary movements on fractures intersecting deposition holes, see Section 8.3.1. A strongly contributing factor to the former could be high groundwater pressures such as might occur during a glaciation.

Addressing the latter failure mode requires a complex evaluation of shear movements for a range of mechanical load situations. For assessing the consequences of such movements the following conditions regarding secondary shear displacements and velocities in fractures at deposition holes have, through the canister design premises and the design analysis of the canister (see above) been established:

$d_{shear} < 5$ cm

$v_{Shear} < 1$ m/s.

R4. Provide favourable thermal conditions

The safety evaluation is simplified if water in the various components of the repository does not freeze. It is, however, not a global requirement that freezing does not occur. For example, freezing is part of the expected evolution for the groundwater down to typically 100 – 200 m depth for permafrost conditions, during which also the closure in the access shaft and ramp is expected to freeze.

The temperature at which the groundwater freezes at repository depth is determined by the groundwater composition and the hydrostatic pressure. Groundwater freezing *per se* is, however, in general not necessarily negative for long-term safety. An exception concerns the possibility of long-term buffer erosion to the extent that a water-filled cavity forms in the deposition hole. The freezing of water in such a cavity could lead to an increased pressure in the canister. The freezing temperature of distilled water at a hydrostatic pressure corresponding to a repository depth of 450 m is −0.3°C.

As mentioned in Section 8.3.2, the safety function indicator for buffer freezing is −4°C. The rock temperature at repository depth should thus not fall below this value.

Furthermore, as also mentioned above, the analyses of the mechanical impact of shear movements is carried out for material properties valid for temperatures down to 0°C, which is thus set as another indicator criterion, that it is necessary to evaluate in the context of consequences of fracture shear movements.

Additional considerations

The overall safety functions for the host rock, expressed as providing favourable chemical, mechanical, hydrological and thermal conditions, are general in nature. The safety function indicators and, where applicable, indicator criteria, to a large extent stem from canister or buffer processes, except for the indicators related to hydrogeology and transport, which are derived from intrinsic host rock properties.

A scrutiny of the geosphere processes is thus not expected to generate additional safety functions to the general categories already identified since all geosphere processes belong to one of these categories and also since the geosphere processes themselves do not, in general, provide indicators and indicator criteria. This was also the result of a brief scrutiny of the geosphere processes.

Safety functions relating to the prevention of future human actions (FHA) with detrimental effects on the host rock, and in particular intrusion into the repository were considered. However, properties of the host rock related to the likelihood of future human actions (mineral or thermal energy resources) and to the efforts/resources required for an intrusion of the repository (depth, sealing) are considered in the siting and design of the repository and are not factors that change over time in the long-term safety assessment. These factors are included in the analysis of FHA scenarios, but are not seen as suitable for the formulation of safety functions.

8.3.5 Summary of safety functions related to containment

The safety function, and associated indicators and criteria derived in the preceding sections are summarised in Figure 8-2. Table 8-1 gives a summary of the margins to appropriate functioning for the safety function indicator criteria for containment, see also Section 8.2, subheading "Approach to margins".

8.4 Safety functions for retardation

Several of the above safety functions and associated indicators and criteria are related only to the containment properties of the system. This is particularly the case for the three canister functions Can1–Can3.

Should a canister be breached, a number of additional phenomena and processes related to the release and transport of radionuclides, i.e. relating to the retarding function of the system, become relevant.

Should a canister failure occur, release limitation and retardation is provided by functions, function indicators and, where applicable, function indicator criteria according to the following. All defined entities are summarised in Figure 8-3 at the end of this section.

8.4.1 Fuel

F1. Contain radionuclides

The fuel matrix cannot be controlled in the design of the repository. Nevertheless, the fuel types to be deposited have a matrix structure that is very stable in a repository environment and therefore provide an important function by containing radionuclides. The fuel conversion rate is a suitable indicator for this safety function. The conversion rate is less for reducing conditions, hence providing reducing conditions is a safety function of the host rock also from the point-of-view of retardation.

Also the structural metal parts of the fuel elements contain radionuclides. The corrosion rate of these metals is, therefore, also an indicator for the containment function of the fuel.

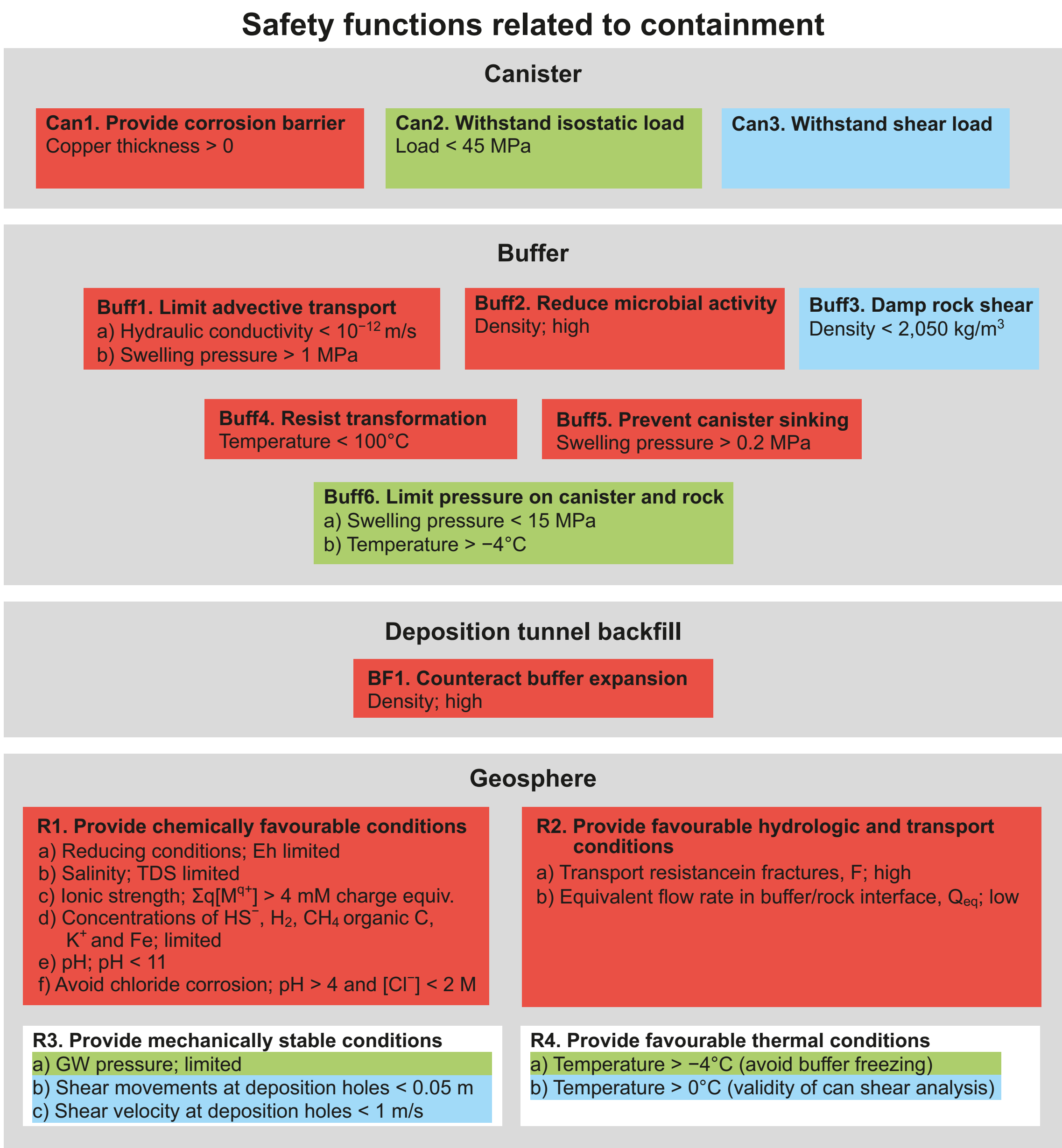


Figure 8-2. *Safety functions (bold), safety function indicators and safety function indicator criteria related to containment. When quantitative criteria cannot be given, terms like "high", "low" and "limited" are used to indicate favourable values of the safety function indicators. The colour coding shows how the functions contribute to the canister safety functions Can1 (red), Can2 (green) and Can3 (blue).*

Table 8-1. Summary of margins for safety function indicator criteria for containment.

Indicator	Criterion	Notes on margin
Minimum copper coverage	0	No margin, for trivial reasons.
Isostatic load on canister	45 MPa	Criterion from design premises. Shown in design analysis to be fulfilled with a margin that could possibly be considerable (up to roughly 100 MPa) for global collapse. See further /Raiko et al. 2010/.
Shear		
Fracture shear distance Maximum buffer density Shear velocity Minimum temperature	5 cm 2,050 kg/m^3 1 m/s 0°C	Set of criteria from design premises. Shown in design analysis to be fulfilled, but 5 cm shear is without a margin to calculated failure for the most unfavourable fracture locations and angles with the maximum buffer density. For buffer densities obtained in the evaluation of the reference design, the margin may be considerable in many situations, although these were not fully evaluated in the design analysis. See further /Raiko et al. 2010/.
Buffer hydraulic conductivity	10^{-12} m/s	The margin is related to the hydraulic gradient and the diffusivity of species in question. The margin is considerable.
Buffer swelling pressure	1 MPa	As the swelling pressure drops, the possibility for pathway formation in the buffer increases. There is an effect of the hydraulic gradient and possibly of salinity. Laboratory samples show piping at ~60 kPa, i.e. the margin to observed malfunction is considerable.
Buffer maximum temperature	100°C	The extent of mineral transformations in the buffer is related to both the temperature and to the duration of the thermal pulse. Since the duration of the thermal pulse is short (on a geological timescale) the margin is considerable.
Buffer maximum swelling pressure (to limit isostatic load on canister)	15 MPa	Contributes to isostatic load on canister. See above for a discussion of margins on limits for isostatic load.
Buffer freezing temperature	−4°C	The freezing point of compacted bentonite depends strongly on density. For the lowest accepted buffer density the freezing point is c. –4°C and for nominal buffer density, the freezing point is lower than –6°C as shown both experimentally and theoretically /Birgersson et al. 2010/. For a given temperature drop below the freezing point, only a portion of the water in the bentonite turns into ice, i.e. any possible pressure build-up occurs gradually with decreasing temperature and will be less than 13.5 MPa/°C, which defines the phase boundary between ice and liquid bulk water. Furthermore, the lowering of the freezing point of water in the buffer will have additional contributions from the hydrostatic pressure at repository depth as well as from any dissolved salts in the ground water.
Buffer minimum swelling pressure (to prevent canister sinking)	0.2 MPa	Modelling has been done for swelling pressures down to 80 kPa, a value for which the consequences of canister sinking could be considered to be acceptable.
Groundwater ionic strength (avoid buffer erosion)	$q[M^{q+}] > 4$ mM	The experimental margin on this value is a factor of two. However, there could be other factors that could limit erosion.
Groundwater pH	< 11	The value is a practical limit. The duration of the conditions with increased pH and mass balances of involved reactions need to be evaluated when consequences are analysed.
Groundwater pH and chloride concentration (to avoid chloride assisted corrosion of copper)	pH > 4 and $[Cl^-]^{GW} < 2$ M	No margin to onset of chloride assisted corrosion, but considerable margin to conditions ever expected in groundwater at repository depth at Forsmark.

F2. Precipitation

Many of the most hazardous radionuclides have limited solubilities in a repository environment, hereby providing an important limitation on radionuclide releases from a failed canister. The elemental solubilities are suitable indicators for this safety function.

Many elemental solubilities are lower for reducing conditions, hence providing reducing conditions is a safety function of the host rock also in this respect.

F3. Avoid criticality

The fuel properties and geometrical arrangement in the canister should be such that criticality is avoided if water should enter a defective canister, but there is no meaningful simple criterion to use for such an evaluation. Qualitatively, the fuel reactivity should be low and it is a design premise that the effective multiplication factor (k_{eff}) for the encapsulated fuel in a water-filled canister should not exceed 0.95 including uncertainties.

Furthermore, the canister insert should have a favourable geometry and material composition with respect to prevention of criticality. This is reflected in geometrical constraints on the canister design and on limitations on the contents of C and Si in the cast iron insert.

8.4.2 Canister

Can4. Provide transport resistance

After a canister failure, water must reach the fuel and radionuclides must be transported through the canister for a release to occur. The nature of the failure will determine whether there are remaining physical hindrances in the canister to transport of water and radionuclides. Although the canister is not designed to provide such transport resistances, they may provide a considerable limitation on release rates, at least over a limited time span after failure.

The delay after failure to an onset of a release (t_{delay}) and a time for expansion of an initial failure to the extent that all transport resistances have been lost (t_{large}) may, for some failure types, be used as indicators for this safety function.

Can5. Avoid fuel criticality

As mentioned above, the canister geometry and material composition should be such that they contribute to the prevention of criticality.

8.4.3 Buffer

Functions in common with containment functions

Also for retardation, the buffer has an important function in limiting advective transport. The criteria on hydraulic conductivity and swelling pressure hence apply also for retention. In order to keep its favourable properties, the buffer should also resist transformation for which there is a criterion on temperature and it should prevent canister sinking that could short-circuit the buffer, ensured through a criterion on swelling pressure.

Buff7. Filter colloids

The buffer should furthermore be dense enough to prevent transport of colloids through it. This requirement is put on the buffer so that fuel colloids should not be able to escape a defective canister. Thereby, the releases of several key radionuclides will be limited by their solubilities. This requirement has led to the following criterion, see the **Buffer, backfill and closure process report**, Section 3.5.4.

$\rho_{Wet}^{Buff} > 1{,}650 \text{ kg/m}^3$

Buff8. Sorb radionuclides

Limited advection in the buffer so that diffusion is the dominant transport mechanism is of primary importance also for radionuclide transport and ensured by the same safety functions as for containment. In addition, the sorption of radionuclides in the buffer may provide a significant limitation on the outward transport of radionuclides. The movement of water through the buffer is strongly limited, through the diffusion dominated transport in an intact buffer. In comparison to water, the transport of radionuclides is further retarded:

- By slower diffusion, which may be caused by a smaller diffusion coefficient in free water and by the electrostatic influence on apparent diffusion-available porosity (anion exclusion).
- By interaction with the clay surface, leading to sorption (expressed as K_d).

The element specific effective diffusion coefficients (D_e) and sorption coefficients (K_d) are suitable indicators for this safety function.

Buff9. Allow gas passage

The buffer should allow gas produced within a potentially damaged canister to escape. The gas transport properties are related to the buffer swelling pressure, where a lower swelling pressure is an advantage, but quantitative limits for favourable buffer function in this respect cannot be formulated at this stage. A limit would be related to the potential damage to the repository from the pressure or release of an overpressurised gas. The buffer issues related to gas transport are dealt with in the assessment of the evolution of a defective canister in Section 13.8.

8.4.4 Deposition tunnel backfill

BF2. Limit advective transport

The backfill should not be a preferred pathway for radionuclide transport. For this to be fulfilled the backfill should have a certain swelling pressure to assure tightness and homogeneity and a limited hydraulic conductivity. The quantitative criteria are the following:

$P_{Swell}^{Backfill} > 0.1$ MPa

and

$k^{Backfill} < 10^{-10}$ m/s

The basis for these criteria is documented in the **Buffer, backfill and closure process report**, Sections 4.3.1 and 4.2.2, respectively.

There is also a requirement that the backfill in the deposition tunnels should not freeze. As for the buffer, the critical temperature is dependent on the swelling pressure. The swelling pressure of the installed backfill according to Section 5.6.3 is around 3 MPa. In the **Buffer, backfill and closure process report**, Section 4.1.2 the following criterion is formulated:

$T^{Backfill} > -2°C$

This is based on pure water and the salinity at the Forsmark site may reduce the freezing temperature further. For the deposition tunnel backfill, this requirement is primarily related to the retardation function, since i) freezing may damage the walls of the tunnel thus creating new transport paths for radionuclides and ii) the transport properties of a frozen and thawed backfill may be less favourable than those before freezing.

BF3. Sorb radionuclides

As for the buffer, sorption of radionuclides in the deposition tunnel backfill may provide a limitation on the outward transport of radionuclides. The sorption coefficients (K_d) are suitable indicators for this safety function.

8.4.5 Geosphere

Functions in common with containment functions

The geosphere safety functions concerning retardation are related to favourable i) chemical and ii) hydrologic and transport conditions and most of the relevant functions and function indicators are the same as for containment:

- A criterion related to the groundwater composition is that of reducing conditions which is of particular importance for maintaining a stable fuel matrix and low solubilities. In addition, indicators relating to ionic strength play a role, mainly in order to ensure favourable conditions for the retarding buffer.
- Regarding hydrologic and transport conditions, requirements on high transport resistances and low equivalent flow rates are common to the corresponding functions for containment.

R2c. Matrix diffusion and sorption

Matrix diffusion and sorption give important contributions to retardation of radionuclides in the geosphere. The element specific effective diffusivities (D_e) and sorption coefficients (K_d) are suitable function indicators.

R2d. Low colloid concentrations

Colloids travelling with the flowing water in fractures may transport radionuclides sorbed to their surfaces, thereby preventing retention of these nuclides by matrix diffusion and sorption. It is therefore favourable if colloid concentrations are low.

8.4.6 Summary of safety functions related to retardation

The safety functions, and associated indicators and criteria derived in the preceding sections are summarised in Figure 8-3. Table 8-2 gives a summary of the margins with which the safety function indicator criteria for containment have been determined.

Table 8-2. Summary of margins for safety function indicator criteria for retardation.

Indicator	Criterion	Notes on margin
Fuel reactivity	$k_{eff} < 0.95$	Established according to principles generally applied for handling of nuclear fuel, see further the **Spent fuel report**.
Backfill hydraulic conductivity	$k < 10^{-10}$ m/s	The criterion is defined to ensure limited overall transport (not to guarantee diffusion). The margin is largely dependent on the characteristics of the rock.
Backfill swelling pressure	$P_{sw} > 0.1$ MPa	Piping has been observed at ~60 kPa, however at gradients much higher than those projected to occur in the repository.
Backfill temperature	> –2°C	According to Table 5-21 the minimum density in a cross-section the backfill will be 1,458 kg/m³. According to Figure 5-19 this would give a swelling pressure of ~2.5 MPa and a critical temperature of −2°C.

Safety functions related to retardation

Fuel – matrix and structural parts

F1. Contain radionuclides
a) Fuel matrix conversion rate; low
b) Metal corrosion rate; low

F2. Precipitation
Elemental solubilities; low

F3. Avoid criticality
Reactivity; $k_{eff} < 0.95$

Canister

Can4. Provide transport resistance
a) t_{delay}; large
b) t_{large}; large

Can5. Avoid fuel criticality
a) Favourable geometry
b) Favourable material composition

Buffer

Buff1. Limit advective transport*
a) Hydraulic conductivity < 10^{-12} m/s
b) Swelling pressure > 1 MPa

Buff4. Resist transformation*
Temperature < 100°C

Buff5. Prevent canister sinking*
Swelling pressure > 0.2 MPa

Buff7. Filter colloids
Density > 1,650 kg/m^3

Buff8. Sorb radionuclides
K_d; high

Buff9. Allow gas passage
Swelling pressure; low

Deposition tunnel backfill

BF2. Limit advective transport
a) Hydraulic conductivity < 10^{-10} m/s
b) Swelling pressure > 0.1 MPa
c) Temperature > −2°C

BF3. Sorb radionuclides
K_d; high

Geosphere

R1. Provide chemically favourable conditions
a) Reducing conditions*; Eh limited
b) Salinity*; TDS limited
c) Ionic strength*; $\Sigma q[M^{q+}] > 4$ mM charge equiv.
d) Concentrations*of $K^+$and Fe; limited
e) pH*; pH < 11

R2. Provide favourable hydrologic and transport conditions
a) Transport resistancein fractures, F; high
b) Equivalent flow rate in buffer/rock interface, Q_{eq}; low*
c) K_d, D_e; high
d) Colloid concentration; low

Figure 8-3. *Safety functions (bold), safety function indicators and safety function indicator criteria related to retardation. When quantitative criteria cannot be given, terms like "high", "low" and "limited" are used to indicate favourable values of the safety function indicators. Safety functions marked with an asterisk (*) apply also to containment, see Figure 8-2.*

8.5 Factors affecting temporal evolution of safety function indicators – FEP chart

As has been mentioned earlier, the general evolution of the repository system, and that of the safety function indicators in particular, is determined by the initial state of the system, by a number of coupled, internal processes and by external influences on the system.

For the purposes of the safety assessment, it is desirable to have an overview of all these factors and their interdependencies. This is, in the SR-Site assessment, obtained through the development of a FEP chart.

A FEP chart contains important initial state properties, important processes, external influences, safety function indicators and the relations between these.

Figure 8-4 shows a FEP chart for a KBS-3 repository, covering factors of importance for containment. The figure shows initial state factors (e.g. the initial copper thickness), processes (e.g. corrosion), safety function indicators (e.g. copper thickness over time) and the safety function indicator criterion (e.g. thickness > 0). Dashed lines indicate influences that occur if a safety function indicator criterion is violated.

The FEP chart contains the following components.

- All variables defined as *containment* related safety function indicators and their criteria, i.e. those given in Figure 8-2. However, all variables that express a component of the groundwater composition are collectively described as "groundwater composition" in the FEP chart, rather than being listed individually.
- Additional variables necessary to describe system evolution and safety, but which are not regarded as safety function indicators, e.g. the porewater pressure of the buffer.
- All identified fuel, canister, buffer and backfill processes related to *containment*, except those which may be neglected according to the **Process reports**. However, some processes in the process tables are lumped into a single process, as indicated in the last columns of the process tables, Table 7-2 to Table 7-5.
- Geosphere processes and variables lumped into a limited number of phenomena that control the system evolution. The lumping is described in the geosphere process table, Table 7-6. The lumping also includes external influences on the system through the division of geosphere process descriptions into those applicable in the temperate, periglacial and glacial climate domains.
- Couplings and influences between the variables and the processes.

The FEP chart is useful in providing an overview of all major safety related factors related to containment, e.g. in the selection (Chapter 11) and analysis (Chapter 12) of scenarios based on safety function indicators.

The biosphere is not represented in the FEP chart since no safety functions are associated with the biosphere. The geosphere is less detailed than the engineered parts of the system since most of the safety function indicator criteria are related to the engineered parts. Factors related purely to retardation are not included. Most retardation factors are, however, important also for containment and are therefore included.

In summary, the FEP chart provides an overview of the relationships between initial state factors, variables, processes and safety function indicators. It aids an expert in analysing the system qualitatively, and is used, in combination with other sources, for scenario selection and analysis in SR-Site.

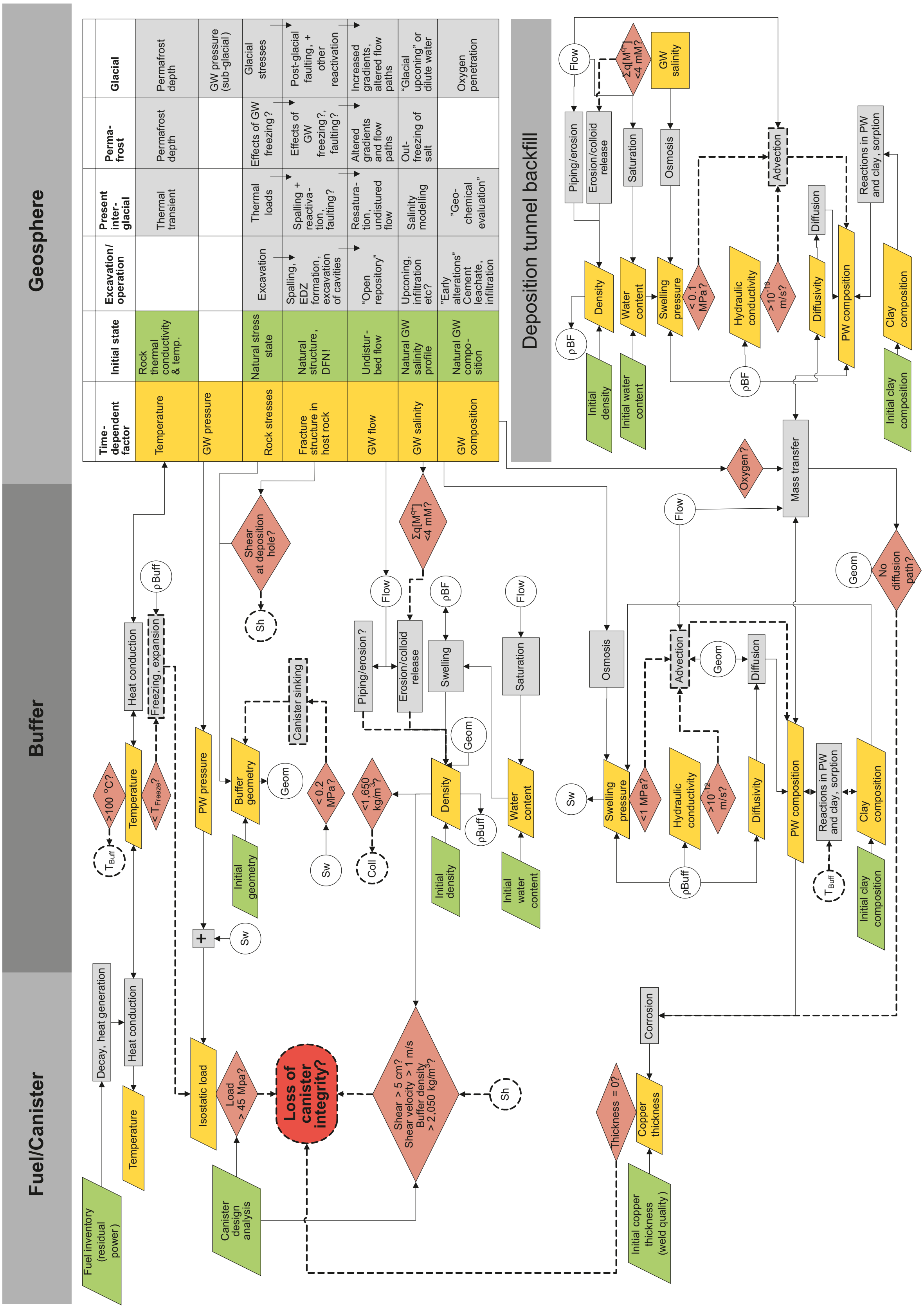


Figure 8-4. The SR-Site FEP chart, covering factors of relevance for containment. Colour coding: Initial state factors, Variables, Processes, Safety function indicators. Solid lines: Influences that always occur. Dashed lines: Influences if there is safety function indicator violation. Circles: Interrupted influence lines (to increase readability).

9 Compilation of input data

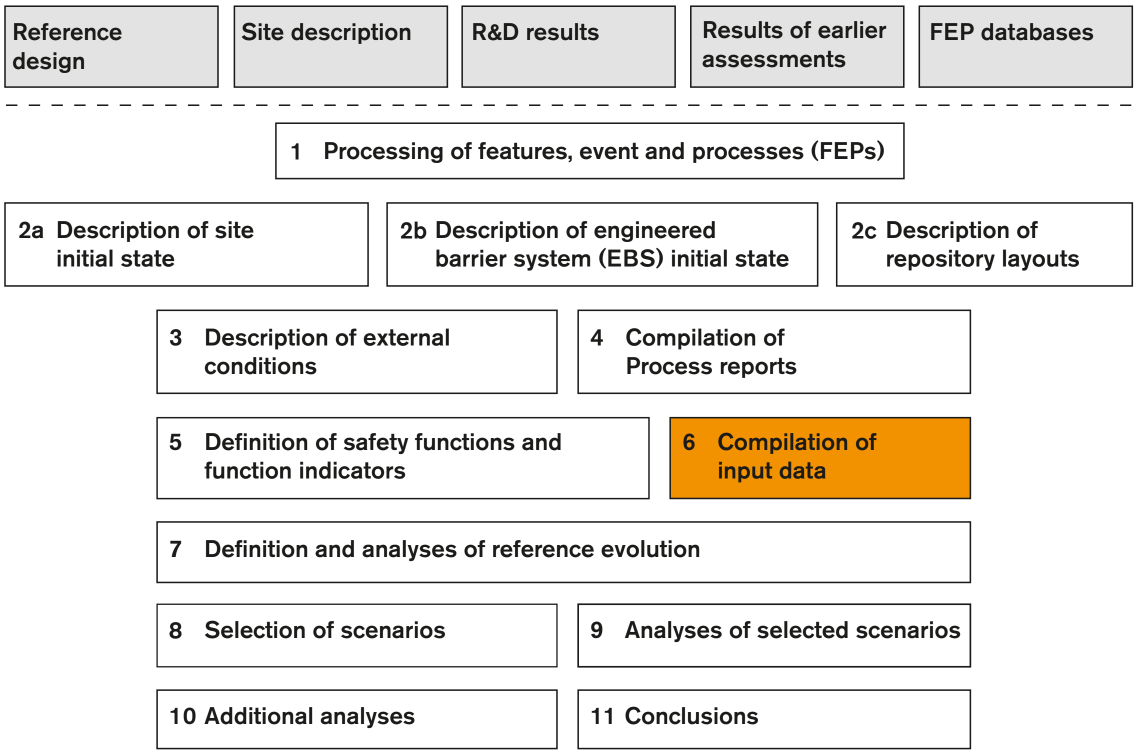


Figure 9-1. *The SR-Site methodology in eleven steps (Section 2.5), with the present step highlighted.*

9.1 Introduction

All input data used in quantitative aspects of the safety assessment have uncertainties. The quality of the results of any calculation in the assessment will, among other factors, depend on the quality of the input data and on the rigor with which input data uncertainties have been handled. A methodological approach for the determination of input data with uncertainties and the subsequent handling of the uncertainties is, therefore, required.

The set of input data parameters for the safety assessment is very large. Some input data uncertainties will have a substantial influence on safety related output uncertainty, whereas others will essentially not influence output uncertainty at all. An example of the latter are transport properties of those radionuclides that never give a significant contribution to the total dose. It is thus appropriate to identify input data to which output is sensitive and use these insights in allocating resources to the determination and, where feasible, reduction of input data uncertainties. It is also important to have a high degree of confidence in the data that are used to conclude that particular nuclides will never contribute to dose.

9.2 Objectives of the SR-Site Data report

The objective of the **Data report** is to compile input data, with uncertainty estimates, for the SR-Site assessment calculations for a wide selection of conditions. Data have been assessed through standardised procedures, adapted to the importance of the data, aiming at identifying the origins of uncertainties and in which the input provided by suppliers is distinguished from judgements made by the assessment team.

9.2.1 Background

Starting with the safety assessment SR 97, a standardised procedure has been employed for the derivation of all input data for the safety assessment. Data relevant to radionuclide migration were presented in the SR 97 Data Report /Andersson 1999/ which was reviewed by the authorities as part of the general SR 97 review /SKI/SSI 2001/. Following SR 97, both SKB /Hedin 2002a, 2003/ and the authorities /Wilmot and Galson 2000, Wilmot et al. 2000, Hora 2002, Hora and Jensen 2002/ have performed investigations relevant to the data derivation process in safety assessment calculations.

The results of these studies and the general development work undertaken were initially reported and applied in the interim version of the SR-Can Data report /SKB 2004a/ in which it was demonstrated how the safety assessment methodology had been developed since SR 97. The Interim version of SR-Can was followed by the SR-Can assessment and the SR-Can Data report /SKB 2006f/ in which site specific data were (naturally) to a much larger extent included. Also the SR-Can Data report /SKB 2006f/ was reviewed by the authorities /Dverstorp and Strömberg 2008/. The general conclusion of that review was that the structure of the Data report and the templates form an appropriate basis for providing the data necessary for the analysis. However, the authorities also concluded that there were no clear distinctions between data that were included in the Data report and data that were reported elsewhere. Neither was there a clear enough distinction between opinions given by the experts supplying data and by the SR-Can team. Moreover, the authorities identified cases where the data selection process was not transparent and also traceability issues on how data were used later on in the assessment.

9.2.2 Instructions for meeting objectives

Based on the authority reviews, and other experience, a specific instruction on "*Supplying data for the SR-Site Data report*" has been developed as part of the Quality assurance plan for the safety assessment SR-Site. This instruction has been written to facilitate methodical and traceable data qualification, where comments made by the authorities form a basis for the improvements in the data qualification methodology.

9.3 Inventory of data

The mapping of safety relevant processes to models, see Section 7.4, yields a set of models that are used to quantify the system evolution, including models for radionuclide transport and risk calculations. The data requirements of these models in principle constitute the input data inventory to be managed in the safety assessment. The importance of the various parameters, however, differs markedly. While data for all the several hundred input parameters must be quality assured, only a limited sub-set are uncertain to an extent critical to the safety evaluation, thus requiring a detailed quantification of uncertainty. These data are identified by sensitivity analyses of calculation results using preliminary input data ranges, often from earlier assessments. A number of calculation end-points regarding both containment and retardation have been considered and sensitivities of these to input parameter uncertainty and natural variability have been determined. Evaluations of calculation end-points and sensitivity analyses regarding general evolution and radionuclide dose and risk provided in the SR-Can project and subsequent analyses, have been used to continuously update the list of data needing rigorous qualification for the SR-Site assessment. The final list for SR-Site is provided in the **Data report**.

In many cases, input data to SR-Site modelling are provided as probabilistic distributions. Experience from previous safety assessments provides information on which part of the distribution contributes most to the radiological risk. Most often it is the upper or lower tail of the data distributions that contributes. This is an aid in the determination of input data, since it informs the analyst that as long as the relevant tail is carefully described, details in the shape of the remaining distribution are less important, or even insignificant.

9.4 Instructions on supplying data

A specific instruction on "*Supplying data for the SR-Site Data Report*" has been developed as a part of the "Quality assurance plan for the safety assessment SR-Site", see Section 2.9.

9.4.1 Suppliers, customers and SR-Site Data report team

The instructions concern two parties, the suppliers and the customer.

- The suppliers supply data that are qualified in the **Data report**. The suppliers are the teams originating the sources of data, as described in the Site descriptive model reports, **Production reports**, and other supporting documents. The author producing text for the **Data report** on behalf of the supplier is called the supplier representative. The supplier representatives should represent the teams, and not rely solely upon their own opinions.
- The customer is in broader terms the SR-Site team that is responsible for performing the SR-Site safety assessment. However, the entire team is not generally involved in each subject area but it is rather embodied by a group of persons with special knowledge and responsibility. The author producing text for the **Data report** on behalf of the SR-Site team is called the customer representative. The customer representatives should represent the SR-Site team, and not rely solely upon their own opinions.

A Data report team, being a subgroup to the SR-Site team administers the **Data report**, and writes the general text in the **Data report** that does not concern specific data. The persons identified as supplier representatives, customer representatives, and members of the Data report team and of the SR-Site team, responsible for different subject areas, are listed among the experts used in the SR-Site assessment, see Section 2.9.4. The customer, supplier, and/or Data report team can in certain cases be represented by the same individual. If so, steps are taken to involve the groups they represent, or other experts, in the handling of the text.

9.4.2 Implementation of the instruction

Upon identification of the supplier and customer representatives for the specific Data report section, and after they accepted their tasks, the instructions for a specific Data report section are normally presented during a meeting between the supplier and customer representatives and a member of the Data report team. The section and the associated data are then developed in steps:

- The customer writes the sections specifying the task and experience from SR-Can etc. according to the instructions.
- After approval by the Data report team these sections are then sent to the supplier, who should then supply the text concerning data and their variability and uncertainties, according to the instructions.
- Several iterations of these steps may occur.
- Finally, part of the SR-Site team and the supplier convene at a meeting to discuss the entire subject area delivery. This meeting, which is formally recorded, is called a Data qualification meeting. At this meeting the data that are recommended for use in the SR-Site safety assessment modelling are formally decided upon. This meeting may also give rise to review comments that should be handled by the supplier or customer.

9.5 Qualification of input data

Data that are closely associated in the context of the SR-Site safety assessment are categorised into one of many different subject areas. For each subject area, the data qualification process comprises a sequence of stages resulting in a text of a standard outline. According to the instructions, some of the sections in these steps are written by the customer, whereas others are written by the supplier. The steps are outlined in the following (for further detail, see the **Data report**). These steps have been applied consistently in qualifying the input data for SR-Site.

Modelling in SR-Site

In this section, the *customer* defines what data are requested from the supplier and gives a brief explanation of how the data for the subject area are used in SR-Site. This information is provided for precisely defining the input data and explaining the context in which the data are to be used. Justification for the use of these models in the assessment is formally given in the **Model summary report**.

Experience from SR-Can

In this section the *customer* gives a brief summary on how the data for the subject area were used in SR-Can. The experience from SR-Can should function as one of the bases for defining the input data required in SR-Site modelling. It should be noted that the teams undertaking the SR-Site and SR-Can safety assessments largely are the same, so transferring experience from SR-Can to SR-Site has not presented any substantial problem. The summary of how the data were used in SR-Can should conform to the following outline;

- modelling in SR-Can,
- conditions for which data were used in SR-Can,
- sensitivity of assessment results in SR-Can,
- alternative modelling in SR-Can,
- correlations used in SR-Can modelling,
- identified limitations of the data used in SR-Can modelling.

More detailed guidance regarding what should be included in the summary in relation to each of these bullets is given in the instructions.

Supplier input on use of data in SR-Site and SR-Can

In this section the *supplier* has the opportunity to comment on the two above sections. The focus for the supplier should be to help the SR-Site team in choosing an appropriate modelling approach, and avoid repeating errors and propagating misconceptions from SR-Can or from earlier safety analyses. Even if a single individual has the roles as both supplier and customer, he or she may still make comment upon the use of data in SR-Site and SR-Can.

Sources of information and documentation of data qualification

This section, as provided by the *supplier*, is devoted to presenting the most important sources of data, as well as categorising different data sets on the basis of their traceability and transparency, and scientific adequacy. Sources of data may include SKB reports, SKB databases, and public domain material. Documents of importance for the data qualification may also consist of SKB internal documents. All underlying documents should be properly cited throughout the **Data report**.

The supplier should categorise data as either qualified data or supporting data. Qualified data have been produced within, and/or in accordance with, the current framework of data qualification, whereas supporting data have been produced outside, and/or in divergence from, the framework. Data may also be categorised as supporting if they are not entirely representative for the Forsmark site or KBS-3 repository. Data taken from peer-reviewed literature have a special position in that they may be considered as qualified even though they are produced outside the SKB framework of data qualification. However, such data are not by necessity categorised as qualified, as they may be non-representative or lack in some other aspect. Data recently produced by SKB, for example in the site investigations, should *a priori* be considered as qualified. However, before the data are formally categorised as qualified, a number of considerations need to be made, as described in detail in the instructions. Data produced outside the data qualification framework should *a priori* be considered as supporting data.

Conditions for which data are supplied

The data for the different subject areas are likely affected by different conditions. Conditions refer to initial conditions, boundary conditions, barrier states, and other circumstances, which potentially may affect the data to be estimated. In the process of qualifying data for subsequent use in safety assessment, an important consideration is to account for the conditions for which data were acquired, and to compare these conditions with those of interest for the safety assessment. The *supplier* has to provide this information.

Conceptual uncertainty

Here, the *supplier* discusses the conceptual uncertainty of the subject area data. Two types of conceptual uncertainty should be discussed. The first concerns how well the data, and the models wherein it is used, represent the physical reality, and the second concerns conceptual uncertainties introduced by the acquisition, interpretation, and refinement of the data.

Data uncertainty due to precision, bias, and representativity

Here, the *supplier* discusses data uncertainty in terms of precision, bias, and representativity. Such uncertainty is associated both with the acquisition of data, for example in the site investigations, and subsequent refinement of data, for example in the site descriptive modelling. Data uncertainty includes neither conceptual uncertainty nor natural variability.

Spatial and temporal variability

Here, this *supplier* deals with the spatial and temporal variability of the data. The natural variability should as far as possible be separated from data uncertainty, discussed in the above section.

Correlations

An appropriate treatment of probabilistic input data requires that any correlations and functional dependencies between those data are identified and quantified. If such exist, they are described by the *supplier*. However, in the extensive work with the FEP database and the **Process reports**, most correlations and functional dependencies between parameters have been identified. Where appropriate, these correlations and functional dependencies have usually also been implemented in the safety assessment models. Correlations and functional dependencies may also have been used when acquiring, interpreting, and refining data. For example, concerning sorption partition coefficients, data have not been acquired for all relevant radionuclides. For elements for which there is a lack of observations, the supplied sorption partition coefficient will have been estimated from data obtained for one or more analogue elements.

Results of supplier's data qualification

In this section the *supplier* presents data that are considered to be appropriate as a basis for selecting input data for use in SR-Site. The general process of reducing data, valuing different data sets, and finally selecting the recommended data for delivery to the SR-Site team should be fully accounted for. The main instructions are as follows.

- If data qualification has already been performed and accounted for in supporting documents, it is sufficient to briefly summarise the process of selecting the delivered data. In other cases the data presented in supporting documents may need reinterpretation and further refinement, in the light of this instruction and/or other information. If so, the process of reinterpretation and data refinement should be fully documented.
- The data sets that the supplier recommends to the SR-Site team should be in the form of single point values, probability distributions, mean or median values with standard deviations, percentiles, and/or ranges, or as otherwise appropriate.
- If no probability distribution can be supplied, but where the data have significant variability and/or uncertainty, the spread in data could instead be described as a range. However, the meaning of the range has to be provided, e.g. does it represent all possible values, all "realistically possible" values or just the more likely values?
- It should be noted that in many cases, at some stage probability distributions must be assigned to numerical data being the input to probabilistic safety assessment modelling. If the supplier feels inadequate to deliver a defined distribution, but for example delivers a best estimate, an upper, and a lower limit for data, it may fall on the SR-Site team to transform such information into probability distributions. This is justified as the SR-Site team may have a better understanding of how the shape of the assigned distributions (especially in their tails) affects the modelling results. The SR-Site team may also, in some cases, have a better understanding of the underlying statistics of the suggested distribution.

- If it is impossible to express the uncertainty by means other than a selection of alternative data sets or by pessimistic assumptions, this is allowed, as long as the supplier clearly documents this together with the motivation for adopting this approach.
- Unless published elsewhere, the numerical values relating to the individual data sets and/or data points should be stored in a database.

For details see the instructions.

Judgements by the SR-Site team

In this section, the *customer* documents the examination of the delivery provided by the supplier, and makes judgment on the data qualification. This text should be produced in close cooperation with persons in the SR-Site team with special knowledge and responsibility. In case of unresolved issues, the final phrasing should be decided upon by the SR-Site team.

In case the SR-Site team needs to suggest probability distributions based on data supplied in the previous sections, this is justified here. Typical choices made are whether to suggest a distribution in the log space or in the normal space, whether to use a truncated or non-truncated distribution, and selecting the shape of the distribution. The base, shape, and truncation of the distribution may affect the calculated risk as well as the influence of extreme values on the assessment results. Therefore it is important to be cautious in selecting such a distribution. For crucial data, where the choice of distribution significantly affects assessment results, the suggested distribution is communicated with those performing the subsequent modelling, with the larger SR-Site team, and with the experts of the supplier team.

Data recommended for use in SR-Site modelling

Based on all the available information, but also on the needs from SR-Site modelling, the SR-Site team makes a final choice of data, often in the form of well-defined probability distributions, including natural variability, data uncertainty and other uncertainty. In some cases where the spread in data is small, single point values may be chosen. The choice should be fully documented and the resulting data should be tabulated. Also guidelines for how to use the data in subsequent modelling should be given, as required. Justifications and guidelines are required to be kept short so that this section mainly contains tabulated data that are easily extractable for the SR-Site safety assessment modelling.

In the process of making the final choice of data, the supplier is consulted one more time in a *data qualification meeting*. Here the formal decision on the data recommended for use in SR-Site modelling is taken, and records of the meeting are made as part of the SKB quality assurance system. The formal decision is acknowledged by those representing the supplier team and those representing the SR-Site team.

9.6 Final control of data used in SR-Site calculations/modelling

The supplied, quality assured input data must also be used in a correct manner in the modelling. Common errors that may appear in the usage of data are that i) the final version of the data set is not used ii) errors and misprints are made in inputting the data in the program code iii) an incorrect data set is used (for example a groundwater composition for temperate instead of for glacial conditions).

After the completion of the **Data report**, it falls upon the modeller to check that the final data sets have been used. In doing this, the modeller also checks that there are no errors or misprints from inputting the data in the code. A specific instruction on "*Final control of data used in SR-Site calculations/modelling*" has been developed as part of the Quality assurance plan for the safety assessment SR-Site.